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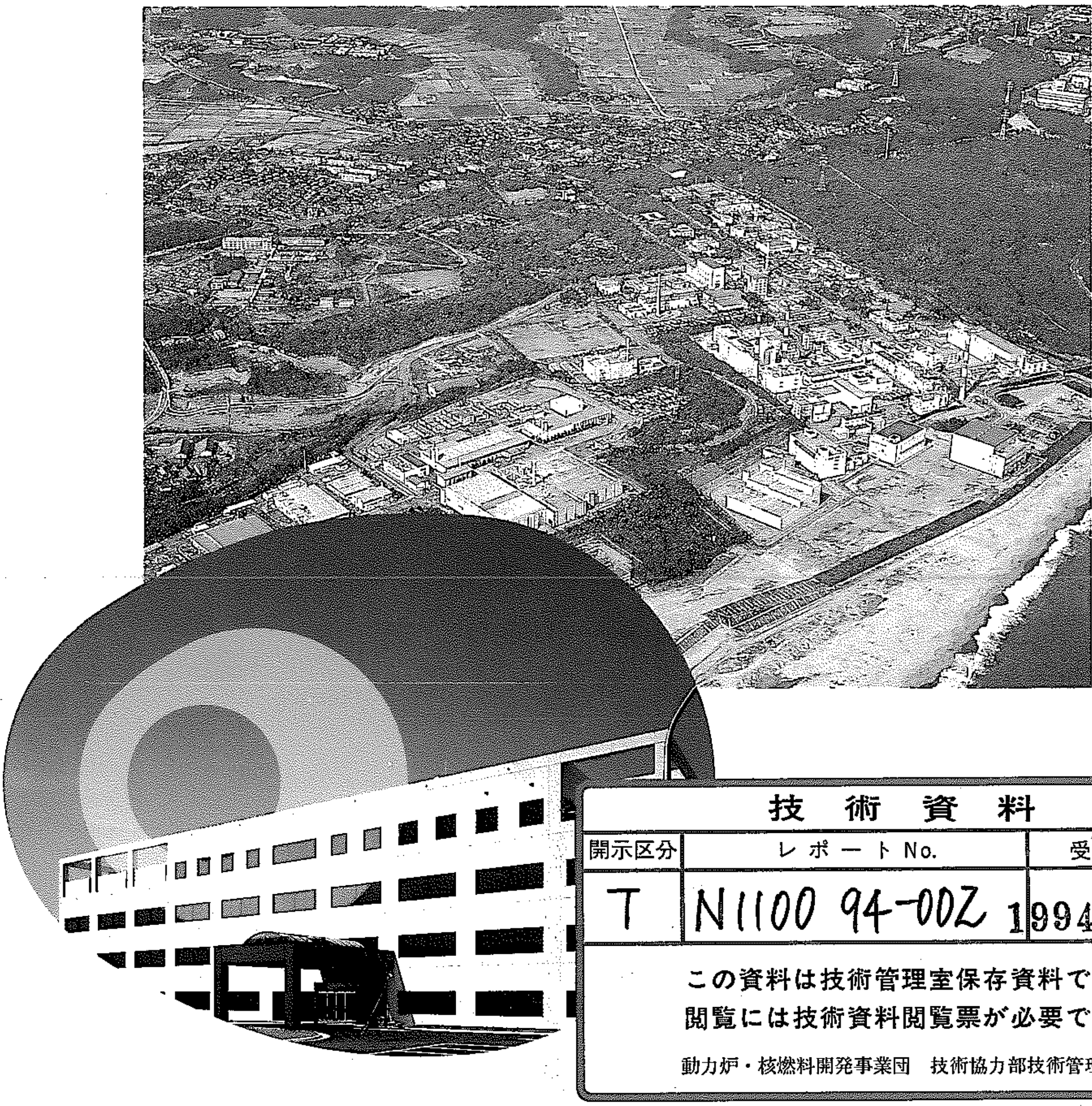
INTERNATIONAL WORKSHOP on RESEARCH & DEVELOPMENT of GEOLOGICAL DISPOSAL

Nov. 15-18, 1993
PNC Tokai, Japan

Proceedings of Plenary Session



POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION

FOREWORD

The International Workshop on Research and Development of Geological Disposal of vitrified HLW and spent fuel was held at Geological Isolation Research Facility in Tokai Works, PNC, Japan on November 15-18th 1993. The Workshop was preceded by the International Symposium on In-situ Experiments at Kamaishi on November 11-12th 1993.

The Workshop consists of a plenary session and technical sessions for discussion among researchers taking direct responsibility for research and development for geological disposal. The plenary session includes presentations on the current status and the future programs for geological disposal of vitrified HLW and spent fuel from relevant countries.

The objective of the technical sessions is to provide an opportunity for an exchange of detailed information and ideas on issues to be discussed in each session. Each session is expected to identify the future direction of R&D approach as an output on the common scientific basis. The closing session summarize the output from each session.

This proceedings includes 8 papers presented in the plenary session. A summary document describing the information presented and conclusion drawn from the technical sessions will be published later including all full papers presented and distributed to the participants.

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International Workshop on Research & Development of Geological Disposal

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Jeremy M. Boak - DOE, USA

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Invited Speaker

Jeremy M. Boak
Yucca Mountain Site Characterization Project, U. S. DOE, USA

Jean-Claude Petit
Direction du Cycle du Combustible, CEA, France

Tonis Papp
Research and Development, SKB, Sweden

Susan M. Sharland
Radwaste Disposal Division, Harwell Laboratory, AEA Technology, UK

Charles McCombie
Science and Technology Division, Nagra, Switzerland

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CURRENT STATUS AND FUTURE PLANS FOR THE U.S. HIGH-LEVEL NUCLEAR WASTE DISPOSAL PROGRAM

Dr. Jeremy M. Boak
Yucca Mountain Site Characterization Project Office
U.S. Department of Energy

Lynn V. Hoffman
Duke Engineering Services, Inc.
CRWMS Management & Operating Contractor

Introduction

Nuclear energy is the second largest source of electric power in the United States. To date, nuclear power plants produced over twenty percent of the nation's electricity. As of August 1991, there were 112 nuclear power reactors in the United States, and two more were being built. By the year 2000, approximately 40,000 metric tons of nuclear waste will be in temporary storage at reactor sites throughout the country. That amount is twice the amount that currently exists.

In order to handle such waste, as well as the added volume to be produced after the year 2000, the U.S. Department of Energy (DOE) is in the process of developing the waste management system that was authorized by the U.S Congress in 1987. The authorized system is illustrated in Figure 1. To summarize, spent nuclear fuel from commercial power reactors will be accepted by the DOE at the reactor sites and transported to a monitored retrievable storage (MRS) facility for temporary storage and preparation for permanent disposal in a geologic repository. From the MRS facility, the spent fuel will be shipped to the repository, where it will be emplaced underground for permanent disposal. The repository may accept spent fuel shipped directly to the repository from nearby reactors. In addition, the waste management system will accept reprocessed high-level radioactive waste.

Most of this waste will be defense high-level waste produced from atomic energy defense activities; the remainder will be commercial high-level waste from the West Valley Demonstration Project. According to current plans, reprocessed high-level waste will be shipped directly to the repository. Also shown in Figure 1 is Federal interim storage, which is an option being evaluated.

The focus of this paper is on that element of the waste management system that pertains to the actual disposal of the waste (i.e., the geologic repository). According to the current schedule, 2010 is the earliest possible date waste could be received at the repository. The goal of the repository program is to dispose of the high-level nuclear waste in a technically sound and environmentally responsible manner. This paper discusses the program the DOE has in place to ensure this goal is achieved.

Background

In 1982, the Nuclear Waste Policy Act (herein referred to as the Act) assigned the U. S. Department of Energy (DOE) responsibility for managing the disposal of spent nuclear fuel and high-level waste. The Act established the Office of Civilian Radioactive Waste Management (OCRWM) to manage the DOE's nuclear waste disposal program. In June 1985, the DOE issued the Mission Plan for the Civilian Radioactive Waste Management Program, which includes the plan for implementing the requirements of the Act. The Act called for the first repository to be selected from nine potentially acceptable candidate sites, which were identified by the DOE in 1983 in the states of Louisiana, Mississippi, Nevada, Texas, Utah, and Washington. In May 1986, the Secretary of Energy recommended three of the sites for characterization, and the President approved the recommendation of the sites at Yucca Mountain, Nevada; Deaf Smith County, Texas; and, in Hanford, Washington.

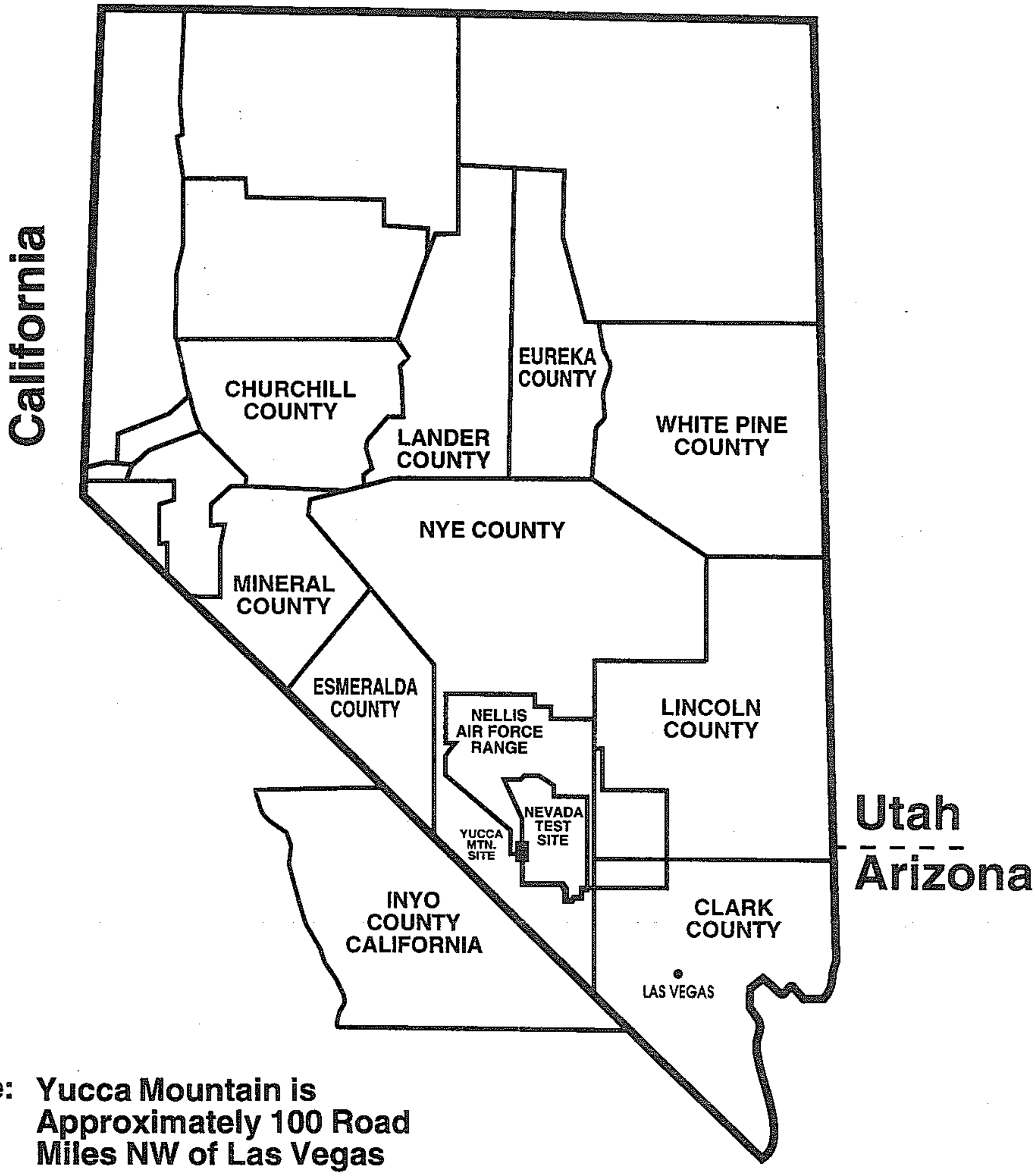
In June 1987, the DOE submitted an amendment to the Mission Plan to Congress, which apprised Congress of significant developments in the waste-management program since the 1985 Mission Plan. At the time the amendment was issued, the preparation of site characterization plans for the three recommended sites was proceeding on schedule. However, at that time, estimates indicated that site characterization would cost approximately $1 billion at each site. It was also apparent that public concern about geologic disposal and the siting process was a principal obstacle in implementing the waste-management program. Because of these and other concerns, the Congress approved legislation to amend the 1982 Nuclear Waste Policy Act. The Nuclear Waste Policy Amendments Act of 1987 limited site characterization to the Yucca Mountain site in Nevada. Figure 1 shows the location of the Yucca Mountain site in Nevada.

The Repository Development Process

Upon completion of site characterization activities at Yucca Mountain, and if the DOE determines that the site is suitable for a repository, the Secretary of Energy will recommend the site to the President. The recommendation will be accompanied by an Environmental Impact Statement, as required by the Nuclear Waste Policy Act and the National Environmental Policy Act. If the recommendation is accepted, the President will submit a recommendation to the Congress. At this time, the State of Nevada may submit a notice of disapproval to the Congress, which would prevent the use of the site unless the Congress passes a joint-resolution of approval. If, at any time during the site-selection process, the DOE determines that the Yucca Mountain site is unsuitable for repository development, then the Secretary of Energy will terminate all site characterization activities and take appropriate steps to notify the Congress and the State of Nevada.

If the Yucca Mountain site is designated for repository development, then the DOE will submit a license application to the Nuclear Regulatory Commission (NRC). The Nuclear Regulatory Commission will review the license application to determine whether or not it will authorize the construction of a repository at the site. This review process is expected to take from three to four years. Construction of the repository will begin following receipt of a construction authorization from the NRC. When the repository is ready for operation, the DOE will submit an updated license application to the NRC for a license to receive and possess radioactive material. As indicated above, the earliest possible date for such operations is 2010.

After the repository is filled to capacity, which is expected to take about 25 years from the start of waste emplacement, it will be kept open for up to an additional 25 years to ensure that the repository is performing as expected and that the emplaced waste need not be retrieved. Following that time, the DOE will submit an application for a license amendment to the NRC that will allow permanent closure of the underground facilities and decommissioning of the surface facilities. The DOE then will apply for a license amendment to terminate the license.



Note: Yucca Mountain is Approximately 100 Road Miles NW of Las Vegas

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Figure 1. Location Map of Proposed Yucca Mountain Repository.

Site characterization

Before the DOE can recommend the Yucca Mountain site to the President for repository development, the DOE must make a determination that the site is suitable. In accordance with the Nuclear Waste Policy Act, the DOE developed guidelines for selecting and developing repositories. These guidelines, know as siting guidelines, are contained in Title 10 Code of Federal Regulations Part 960, and specify conditions that would either qualify or disqualify a site. The postclosure guidelines relate to the ability of the site to contain and isolate waste after the repository is permanently closed. The preclosure guidelines relate to characteristics that could effect the public, the environment, or workers during siting, construction, and operation of the repository before closure. For a site to be considered suitable for development, it must satisfy all of the qualifying conditions and must have no disqualifying conditions.

To demonstrate this, extensive geologic data describing the site must be collected under a program of site characterization. In 1988, the DOE published its Site Characterization Plan (SCP), which describes the activities necessary to determine whether or not the Yucca Mountain site is suitable for a nuclear waste repository. The activities described in this plan consist primarily of surface-based field studies, the construction of an exploratory studies facility, and the tests to be conducted in that facility.

As described in the SCP, surface-based tests consist of geologic, geophysical, hydrologic, geochemical, and other tests and surveys performed both at the land surface and in exploratory boreholes drilled from the surface. The surface-based studies program includes a systematic drilling program to examine the site and the surrounding area to collect samples and data on lithostratigraphy, basic physical properties, fracture characteristics, mineralogy, *in situ* moisture conditions, and other characteristics. The features sampling program investigates the faults and other special features at the site. Other activities include mapping, geophysical surveys, trenching, monitoring, meteorological studies, and laboratory testing.

Surface-based testing supports resolution of technical issues related to regional hydrology, flow and transport, seismic hazard analysis, and volcanism. To date, 232 boreholes have been drilled, 117 trenches have been excavated, and 108 soil pits have been completed for site characterization. From now until 2001, 76 deep and 58 shallow boreholes are scheduled to be completed, as are 25 trenches and from 40 to 50 soil pits. The information obtained from the boreholes is used by many investigators in activities such as gas monitoring for characterizing gaseous releases, and determining rock properties for design and performance assessment. The information is also used to determine stratigraphic boundaries for exploratory studies facility design and for geologic and tectonic history, as well as to monitor water levels for hydrology and tectonic effects. Surface-based trenching assists in the determination of the tectonic history of the Yucca Mountain site.

The underground activities described in the SCP consist of both the testing to be performed in the exploratory studies facility and the associated construction and operations activities necessary to support the testing. These activities include systematic mapping and sampling, tests to characterize processes and conditions, and exploratory drifting. The DOE's mission in developing the exploratory studies facility is to provide access to geologic horizons to evaluate suitability of geologic barriers to isolate nuclear waste. The facility will enable testing to be performed in an underground laboratory. The facility will assist in the determination of site suitability by providing access to the potential repository horizon for inspection and testing and by providing access to the Calico Hills level for inspection and testing. The Calico Hills level, below the potential repository level, is the primary barrier for radionuclide transport because of its sorptive capacity for cationic solutes. Finally, the facility will develop data for use in designing and constructing a potential repository. Figure 2 illustrates the current design of the exploratory studies facility.

Construction of this exploratory studies facility is underway. In September, excavation of the 60 m starter tunnel was completed, and excavation has begun of the first test alcove located 43 m into the starter tunnel. The DOE also recently awarded a contract to purchase an 8.3 m diameter tunnel boring machine that is expected to be delivered in April 1994 and to begin startup in July 1994. The tunnel boring machine will be used for the main drifts and access ramps of the exploratory tunnel facility.

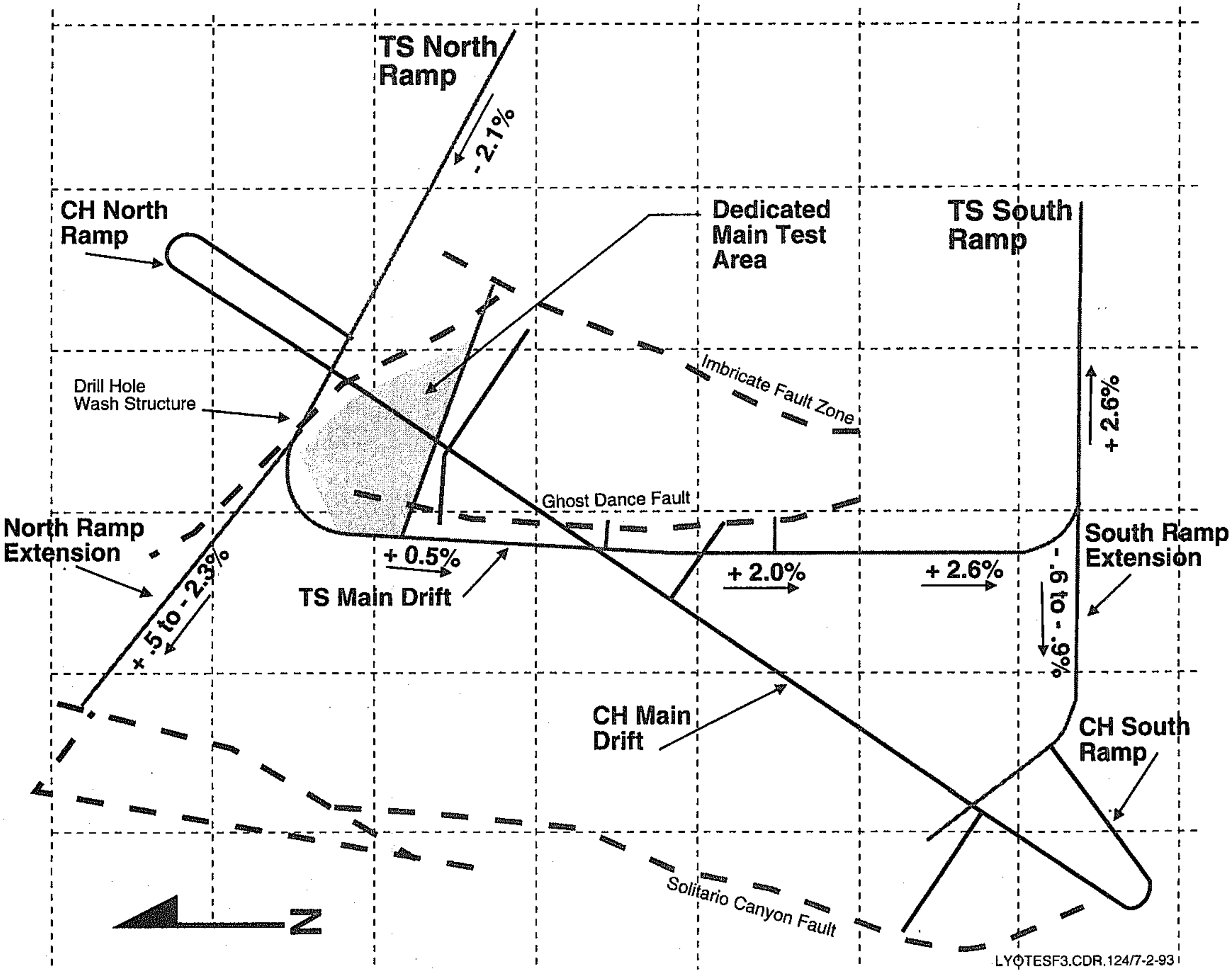

Figure 2. Enhanced ESF Layout: TSw2 and Calico Hills Drifting

Repository conceptual design

According to the 1985 Mission Plan, there are to be four repository design phases: (1) Site Characterization Plan-Conceptual Design; (2) Advanced Conceptual Design; (3) License Application Design; and (4) Final Procurement and Construction Design. The first phase of the repository design process was completed in 1987 with the issuance of the Site Characterization Plan-Conceptual Design Report. The purpose of the conceptual design was to concentrate on the design components that require site-characterization data and to identify the design-related information that must be collected during site characterization. The Yucca Mountain Project is now into the second phase of advanced conceptual design.

Before the advanced conceptual design is completed, many studies will be performed to assist in determining and refining criteria and the concepts to be incorporated into the license application design.

If the Yucca Mountain site is selected as the site for a repository, the mined geologic disposal system surface and underground facilities will be constructed on land owned by the United States government on and adjacent to the Nevada Test Site in southern Nevada (as shown in Figure 1). The proposed geologic repository will consist of surface facilities, underground facilities, and shafts and ramps connecting the surface and the underground facilities. When the repository is prepared for permanent closure, seals will be constructed in the shafts, ramps, and exploratory boreholes.

The objective of the repository is to isolate radioactive materials using natural and engineered barriers. The major characteristics of the proposed repository are that it is designed to accommodate 70,000 metric tons of spent nuclear fuel and waste; there will be approximately 140 km of tunnelling within the repository; the design operational life of the repository is 100 years; the approximate area of the repository is approximately 518 hectares; the depth below the surface averages from 300 to 600 m; and, the minimum distance above the water table is approximately 200 m. Figure 3 illustrates the underground location of the proposed repository within the Yucca Mountain host rock. Figure 4 shows the conceptual layout of the repository ramps and drifts.

In the conceptual design, the underground facility would consist of three parallel main entry drifts and a number of panels, which are areas of solid rock in which the waste would be emplaced. A number of emplacement drifts would be spaced within each emplacement panel in which boreholes would be drilled for waste emplacement. After the waste-emplacement panels have been completely developed, waste emplacement would begin in the first panel.

This process would allow both underground-development and waste-emplacement operations to proceed in parallel and would allow sufficient separation of operations to protect the workers from radiation.

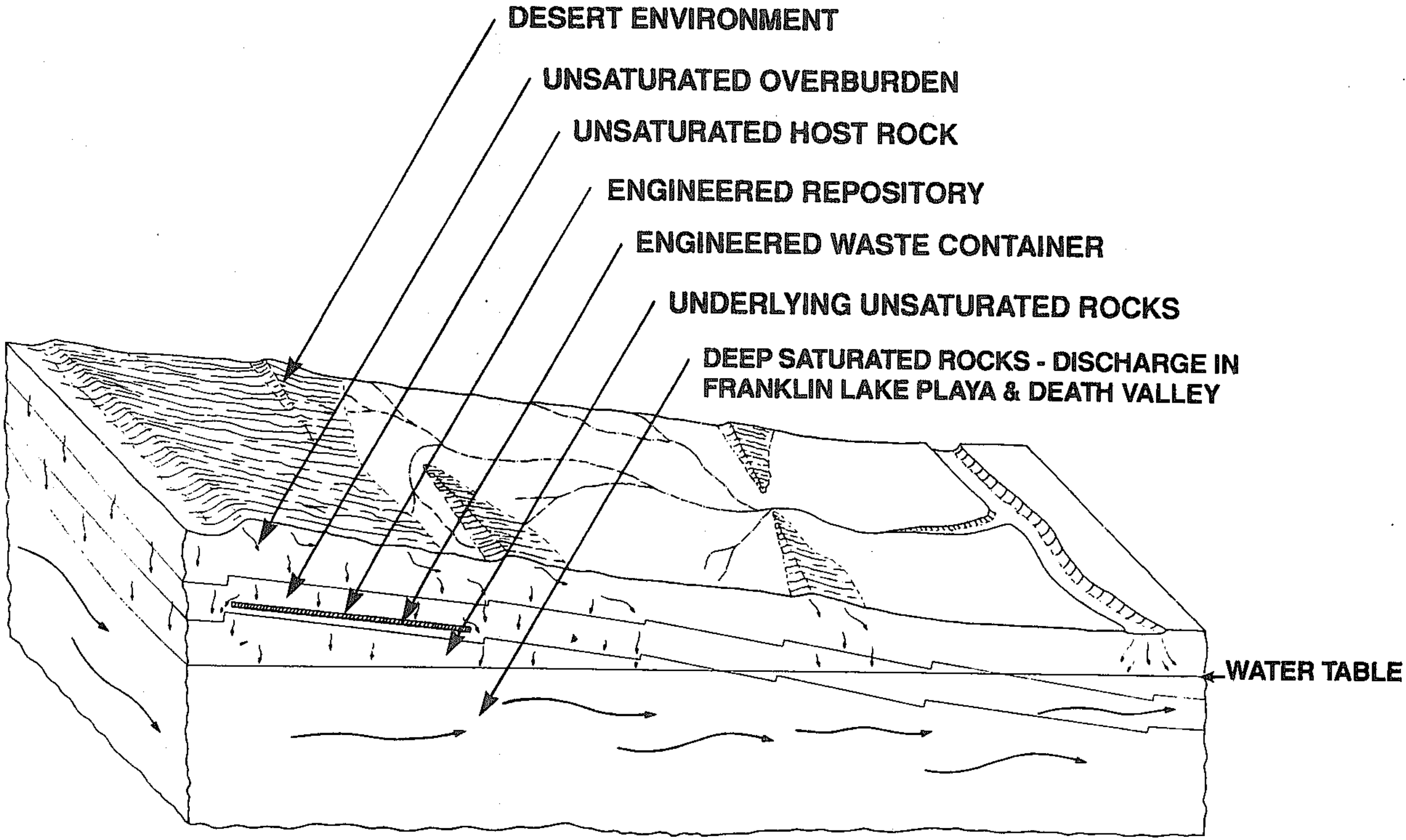

Figure 3. Underground Placement of Proposed Repository Within the Yucca Mountain Host Rock.

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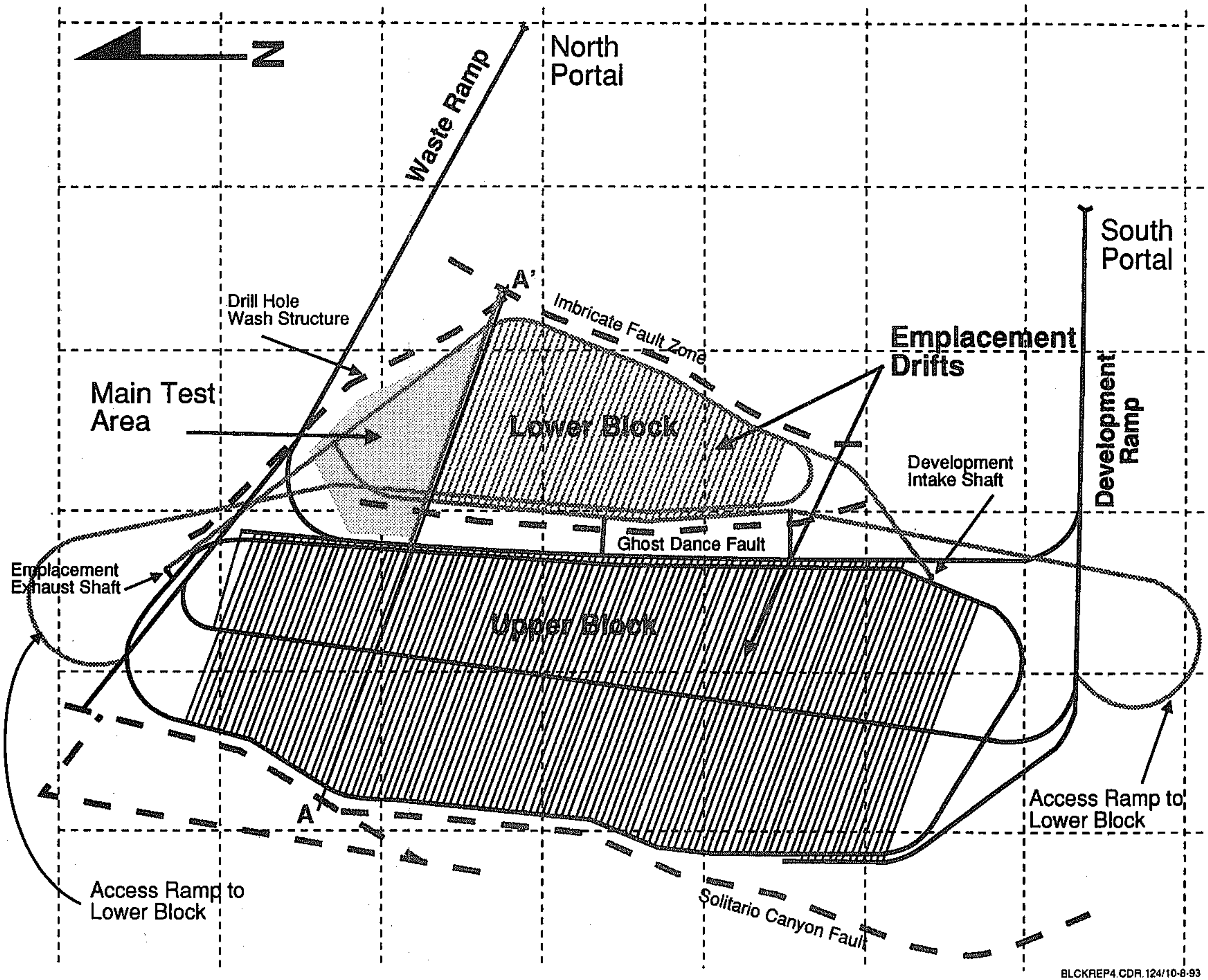

Figure 4. Two Block Repository Concept for Proposed Yucca Mountain Repository.

Waste package and emplacement

The baseline conceptual design in the SCP calls for vertical emplacement of waste. In this waste-emplacement mode, boreholes would be drilled vertically into the floor of the emplacement drifts, and one waste container would be emplaced in each borehole.

The design of the waste package consists of the waste form and a disposal container. The waste package design must meet functional and regulatory requirements including preclosure requirements for radiation protection and for maintaining the option to retrieve the emplaced waste. The postclosure requirements include performance objectives to provide containment for the waste for at least 300 to 1,000 years and for the rate of radionuclide release to be controlled thereafter by the engineered-barrier system. The waste-emplacement environment and the waste-emplacement borehole design will contribute to the postclosure performance of the waste package because it will allow for an air gap between the waste package and the host rock. The design of the waste package will continue to evolve as data are obtained from site characterization and as more detailed phases of design are completed.

The conceptual design calls for the high-level nuclear waste to be commingled in the repository with spent fuel from commercial reactors. The high-level waste from both defense activities and from the West Valley Project is to be in the form of borosilicate glass in metal canisters. The Reference Design for the waste package/engineered barrier system that was documented in the Site Characterization Plan (SCP) described waste packages that would have consisted of a relatively thin-walled stainless steel container with spent nuclear fuel assemblies or a high-level waste glass canister inside. The engineered barrier system would have consisted of these packages inserted into vertical boreholes at the bottom of each drift. A pedestal would separate the bottom of the container from the bottom of the borehole, and an air gap would separate the container wall from the borehole wall. A shielding cap would be placed over each filled borehole. One version of an emplaced waste package fitting this conceptual design is illustrated in Figure 5. Figure 5 also outlines some of the considerations that have to be addressed in evaluating the performance of this type of waste package and emplacement.

This design, as well as a number of more recently developed alternatives, are to be evaluated in the Advanced Conceptual Design (ACD) process, which has commenced. Figure 6 illustrates a version of one of these recently proposed alternative designs, and lists its functional components. The illustration shows a multi-barrier waste package emplaced in a drift rather than a vertical borehole. The multiple barriers include an outer corrosion-allowance material and an inner corrosion-resistant barrier. Corrosion-allowance materials are those that are subject to known general corrosion processes that proceed at predictable rates. Corrosion-resistant materials, on the other hand, resist the general corrosion processes, but are subject to other processes such as pitting or stress-corrosion-cracking. The use of a combination of metallic and/or ceramic materials is also under consideration. The resulting multi-barrier waste package would be robust in the sense that failure would be delayed, and the package would be fairly tolerant of the emplacement environment.

The emplacement mode options are borehole and in-drift. Borehole emplacement requires relatively small packages that can be fitted into vertical or horizontal holes drilled into the bottoms or sides of drifts. Borehole emplacement requires that spacing between containers be determined at the time the boreholes are drilled. Drift emplacement involves placing the container on the drift floor. With proper design and equipment, larger, more robust packages may be emplaced in this mode, and spacing between containers may be adjusted, prior to closure, to optimize the thermal environment.

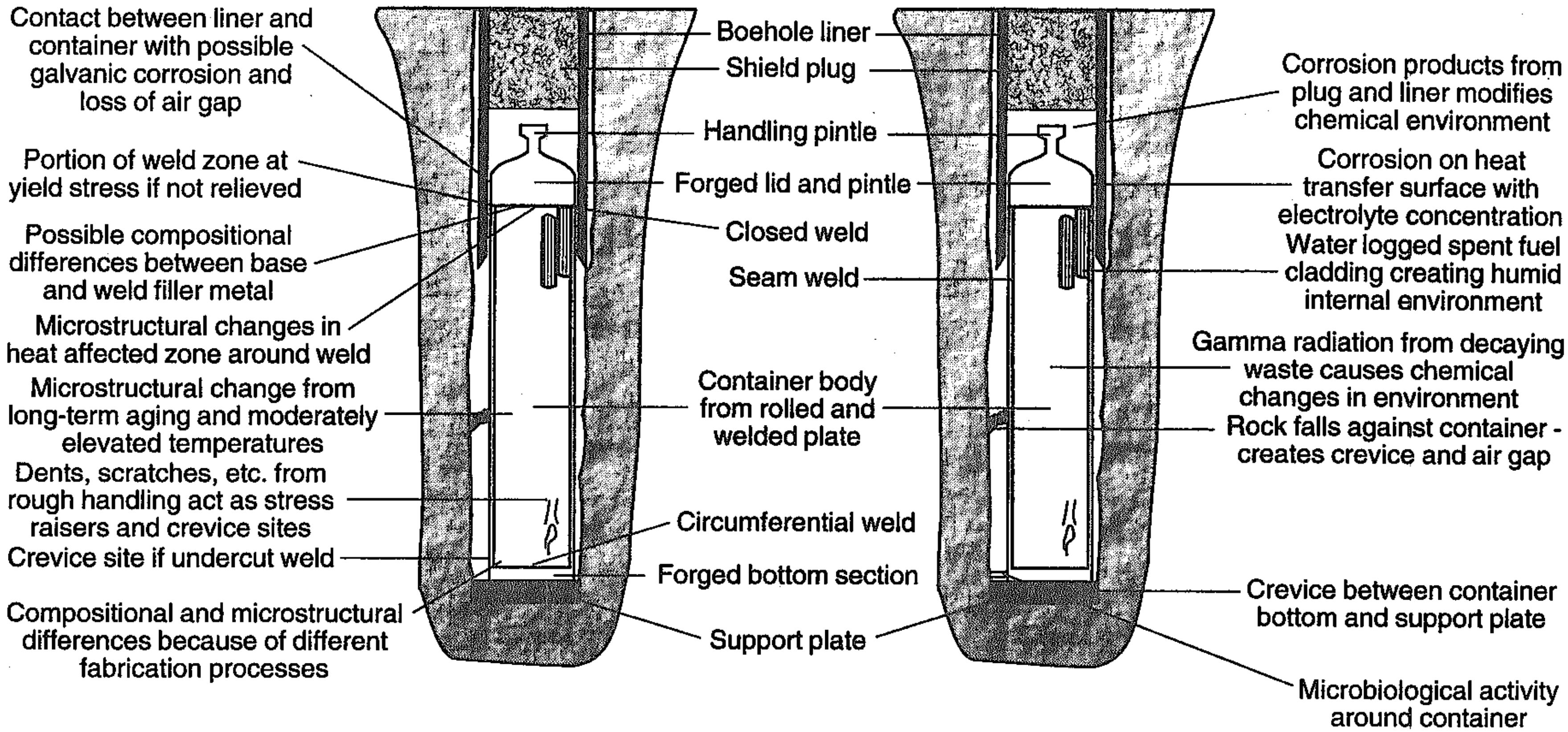

Figure 5. Metallurgical, Mechanical, Chemical and Environmental Container Performance Considerations.

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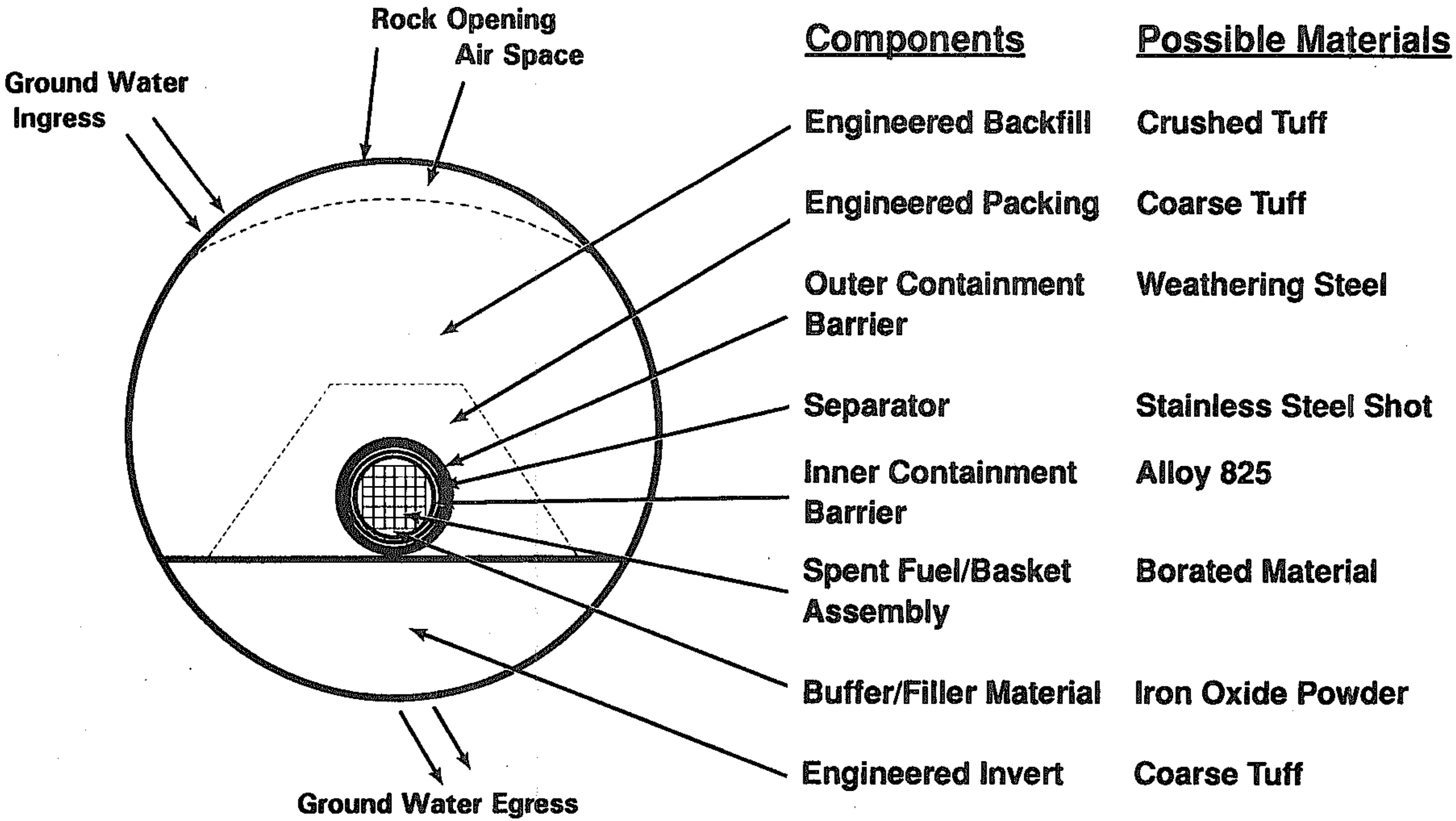

Figure 6. Waste Package/Engineered Barrier System Performance Assessment of ONE Design Concept.

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Systems integration and performance assessment

The DOE has a process that integrates all elements of site characterization activities. Data and information gathered in site-investigation and testing programs are gathered and fed to other program elements. The data are analyzed using a variety of sophisticated computer models and suites of models throughout the program.

To assess the performance of a high-level radioactive waste repository, calculations can be categorized into the pre- and postclosure periods. As stated previously, the preclosure period covers the period of repository operation, closure, and decommissioning. The calculations for this period must demonstrate compliance with the radiation exposure and radioactive material release limits for the unrestricted area and the exposure limits for repository workers during the period of waste emplacement until final closure and decommissioning. For the postclosure period, performance assessment calculations must cover the period following permanent closure of the repository. Calculated releases from the repository must meet the limits specified by the Nuclear Regulatory Commission in Title 10 Code of Federal Regulations Part 60.

The DOE's performance assessment program is composed of a suite of computer models that can be visualized as a pyramid with the most complex process models at the bottom of the pyramid leading upward to the simplest process models and the system models at the top of the pyramid. The process-based models are usually multi-dimensional with coupled process and interaction for understanding the several and various phenomena occurring throughout the repository environment. The system models are abstracted and combined process component simulators, often with stochastic input and multiple realization outputs.

The DOE performed a total system performance assessment in 1991 in which release pathways were modeled. The assessment included aqueous and gaseous flow and transport for a range of potential climatic conditions, as well as modeling volcanic and human disruptions of the repository.

The results of the 1991 total system performance assessment helped identify important priorities for the next iteration of total system assessment. Recommended improvements included: additional sensitivity studies, particularly on parameters for flow and transport calculations, incorporation of more sophisticated gas flow models and more realistic source terms, evaluation of the effects on flow and transport of repository thermal history and of spatial heterogeneity of hydrologic properties, refinement of fast-path flow models, incorporation of tectonic disturbances, development of a better model for saturated zone flow, and incorporation of new site data. These technical enhancements are intended to support programmatic priorities for selection of repository thermal loads and other design parameters, as well as for iterative evaluation of aspects of site suitability to direct site characterization.

The second total system performance assessment is in progress. This second total system analysis is addressing whether the thermal loads proposed in the DOE's Site Characterization Plan were optimal, or if lower (sub-boiling) or higher (extended dry-out) thermal loads would result in better performance and/or more defensible evaluations of site performance. At the same time, industry interest had increased in the concept of multi-purpose containers for the spent fuel waste form. This interest dovetailed with re-evaluation of the waste package concept to include larger packages emplaced in the repository drifts (which may be required to achieve high thermal loads), rather than in boreholes in the drift floor (the SCP concept). This introduction of varying thermal conditions demanded that a range of chemical, thermal, and hydrologic effects on waste form and waste package degradation be addressed in more detail than in previous analyses.

Waste package performance assessments are being performed in support of the advanced Conceptual Design and the total system performance assessment activities. The current total system performance assessment activity is using source term developed by Lawrence Livermore National Laboratory. The source term model being used is an abstraction of the more detailed modeling being done by Livermore to address engineered system performance criteria. The Livermore source-term code, used to support these total system assessments, is the integrating-level code YMIM. It is also being used in addressing aspects of design, test-evaluation, and regulatory questions.

For assessments that support site suitability, environmental impact, or licensing decisions, however, a more sophisticated code is needed at the mechanistic process level of detail. For this more detailed model, one of the existing program codes' executive frameworks has been recommended as a starting point. The recommended framework is to be augmented with compatible versions of the best modules and features of other existing codes, and the modules and features identified as needing development. The AREST code framework has been selected for further development because it is the most advanced and versatile: the code has been benchmarked, in part, against the Canadian VAULT model. AREST has been used to evaluate spent fuel as a nuclear waste form. It has also been used to calculate the source-term for two earlier total-system analyses. A personal computer version is currently available. Finally, a new version, with improved glass waste capabilities, was developed for the Power Reactor and Nuclear Fuel Development Corporation of Japan.

R&D DEVELOPMENTS ON THE DISPOSAL OF RADIOACTIVE WASTES RECENTLY CARRIED OUT AT CEA (FRANCE)

Jean-Claude Petit*

I. INTRODUCTION

Since the law of december 31, 1991 voted by the French Parlement, disposal in geological formations is not the sole option considered for the management of radioactive wastes (JO, 1992). Scientists and engineers, notably within the 'Commissariat à l'Energie Atomique (CEA)', must investigate in parallel two other options, namely i. advanced separation of long-lived radioisotopes and transmutation, and ii. improvement of radioactive waste matrices and long-term storage. Still, the geological disposal of radioactive wastes, which has generated within the last decade an immense amount of R&D, remains a major proposed solution to the long-term management of transuranic and high-level wastes (French categories B and C). In France, this topic is under the responsability of the 'Agence Nationale pour la Gestion des Déchets Radioactifs (ANDRA)', a body now independant from CEA. According to the law, investigating this option should involve the building and exploitation of two underground laboratories and the consideration of both reversible and irreversible disposal. The corresponding investigations are carried out in cooperation with a wide range of research institutions, laboratories, engineering schools and universities, amongst which CEA plays a particular rôle.

In this paper, we will describe some of the R&D carried out within CEA, avoiding the industrial and (strictly speaking) engineering aspects of the issues dealt with. Rather, we will illustrate major points of our basic research, underlining in particular methodological considerations. This paper does not pretend to give an overview of all research activities associated with the disposal of radioactive wastes in our institution. In addition, we do not tackle at all the R&D concerning the two other topics of the law of 1991, namely the advanced separation of long-lived radioisotopes and transmutation (SPIN programme), and the improvement of radioactive

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waste matrices and long-term storage. Readers interested by SPIN should refer for instance to Bourgeois (1992). To our knowledge, no general paper in the open literature is yet available for the second topic. Finally, due to the character of a review of CEA programmes of this paper, quotations will essentially be those of our institution.

II. METHODOLOGICAL CONSIDERATIONS

An important part of our present work can be considered as methodological, since no concept (as already said) and no specific site is indeed specified. In effect, it is well recognised that the extrapolation of the behaviour of a radioactive waste repository (every of its components but also as a whole system), over long periods of time not directly accessible to the laboratory, is a difficult issue. It certainly implies modelling of the system, notably the transport and dispersion of radionuclides through the involved barriers, but the basic question remains : how can we build a reliable model when its long-term predictions cannot, properly speaking, be checked (we purposedly avoid here the word 'validated' which has a too technically-limited meaning) ? In addition, it may be important that this model be easily demonstrable for pertinence for more that a handful group of highly specialised experts. In effect, the building of a consensus amongst the wider scientific community is probably important for public acceptance. Moreover, the possible direct understanding of the public and of the decision-makers of what engineers and scientists do in this field (and not only through their delegation of competence to a very limited number of experts) must not be neglected.

We think that this issue can be tackled by a variety of complementary experimental and observational approaches which enable us i. to identify basic mechanisms, ii. to formalize relevant processes in a model and iii. to check its predictions at **all scales of space, time and complexity** involved in the issue. This general principle is, of course, not so easy to apply, but four directions of research have been explored during the past decade :

- laboratory experiments,
- mocks up,
- in situ experimentations and measurements,
- natural analogues.

We will now give our feeling about the respective rôles of these four experimental and observational tools which should converge in a model (or models).

Laboratory experiments

In such experiments, the relevant conditions are (or, at least, should be) carefully constrained. Only few, well-identified, parameters can be varied in a controlled manner. In principle, the experiment is designed in such a way that involved processes are decoupled as much as possible, thus allowing the identification of the basic active mechanisms. Usually, such experiments are strongly influenced by a priori theoretical considerations. From the viewpoint of the scales of space, time and complexity, laboratory experiments are supposed to be the simplest tools.

Mocks up

Such experimental devices are new in this field but are extensively used in many engineering activities. Indeed, apart from some pioneering works in geophysics, in relation to simulations of plate tectonic effects, it has not been developed either in the wider earth sciences. But interest in this particular approach is steadily increasing. To our knowledge, in our field, only canadians and, above all, japaneses (through their major ENTRY project) have made advances in this area. Although time remains that of the laboratory (i.e. extremely short compared to that involved in the performance assessment of a radioactive waste repository), the space scale and complexity of the investigated system can be notably increased but still in a well-constrained way and with the possibility of a constant monitoring of a variety of parameters. Some coupling between major processes can be investigated. In principle, mocks up should play an intermediate rôle between laboratory and in situ experiments.

Another interesting rôle of mocks up is that they can help to 'see and touch' the diverse modules of a repository, in particular the engineered components of the near-field, and to show to non technical people how this combination of complementary barriers can insure confinement of radioactive wastes. In countries where this approach is developing, such a possibility is considered an important aspect of the mocks up projects.

In situ experimentations and measurements

Experimentions in situ, or more specifically in an underground laboratory, can be considered as 'hybrides' between the well-controlled laboratory conditions (see above) and the complex natural system. The space scale and complexities are still increased, compared to a mock up, but the time scale remains quite short. It allows the use of actual ingredients such as the in situ rock or the deep fluids in realistic conditions (e.g. temperature, redox potential, etc.). But, underground laboratories can also be used, and likely above all, for the characterization of the deep-seated host medium: rock and fluids, heterogeneities (e.g. fractures in crystalline rocks), etc. Such in situ measurements should be considered as reference data. The results of these activities are often of the 'black box' type because many processes can be coupled, relevant parameters not always measurable and thus mechanisms often not directly accessible. Nevertheless, they are indispensable because they are as close as possible to both the methodological conditions necessary for the full application of the scientific method (including model validation) and realistic conditions.

Natural analogues

Analogues allow us to investigate 'how things are actually going on in Nature' at proper space and time scales as well as degree of complexity (Petit, 1992a). They are (to use J.-Ch. Peyrus' saying) a 'phenomenological memory' of natural processes. It is sometimes misleadingly considered that the main purpose of natural analogues is to help validate models developed in the laboratory (even, in some cases, proposed by a theoretist without much experimental support !). Although natural analogues can be (and have indeed been in the past) used in such a way, where initial and boundary conditions could be satisfactorily controlled by scientists, it is not the principal interest of this approach. Our opinion is that analogues are essentially useful as heuristic tool for the in-depth understanding of complex natural phenomena, for the identification of processes and mechanisms, notably those active on the very long-term, and for the building of a robust (realistic and trustful) model describing the overall behaviour of the system under investigation. They play an irreplaceable rôle in constantly questioning scientists and engineers as to the pertinence of their understanding and formalization of issues.

Depending on the different topics, the advancement of our understanding (as well as the corresponding experimentations and models) goes to very simplified laboratory experiments to complex in situ measurements and natural analogues studies. The aim of our research is to safely move from the simple to the complex, in increasing the space and time scales at which we are able to understand, simulate (where appropriate), model and validate the processes active in the investigated system.

III. DISPOSING OF RADIOACTIVE WASTES IN GEOLOGICAL FORMATIONS

Although not any more the sole option for both alpha wastes and high-level wastes (categories B and C in France), the geological disposal has nevertheless generated a huge amount of research, notably in recent years, and, therefore in this article, this option is stressed. A major change for engineers and scientists was induced by the law of 1991. Indeed, although our safety perspective remains, as in most countries, a multi-barriers system insuring the confinement of radioactivity, no practical concept (i.e. with specified engineered components) is now available in France. The law gives us time to reconsider the entire problem, to investigate and test, under the auspices of ANDRA, various ideas for their pertinence and feasibility. But, the concept itself is still at an early stage of 'reconstruction'. Hence, the work of the CEA scientists and engineers is actually very open.

The programme conducted in the author's team can be schematically described as addressing two categories of issues:

- geological and geochemical issues, where the characteristics of the host rock and the chemical evolution of the different parts of a repository are considered,

- materials science issues, where the problems associated with the definition and development of adequate radioactive waste matrices, technological barriers and engineered structures are dealt with.

We will successively highlight recent developments in these two fields.

Geological and geochemical issues

Our approach is based on the consideration that a repository can essentially be described as a 'system' composed of different parts (or modules in the model) which may interact on each other. As usual, we arbitrarily divide the repository into its 'source term', near-field and far-field components. In addition, this system can be considered as a quadri-phasic one made up of solid(s), solution(s), particulate matter(s) and gas(es). We expect that interfaces between these phases may be of particular importance.

The main objective is thus to investigate the short-term and long-term geochemical evolution of this system, in each of its parts, as well as to understand and predict the consequences that it may have on the migration of the main radionuclides of interest.

One important aspect is the variable physico-chemical charateristics of the groundwater which will percolate the host rock, from the surface to the greatest depths. Experts are quite certain that this evolved groundwater will eventually reach the repository, despite the technological barriers which should be interposed between the wastes and the host rock. This issue is particularly important because it will impact the 'agressiveness' of the waters with respect to the waste matrices and affect the migration properties of radionuclides which essentially depend on their chemical form in solution (speciation). Indeed, there is a strong consensus within the scientific community on the idea that solutions will be the main vector of radionuclides transport towards the biosphere.

During the past few years, we have done great efforts in trying to understand and model the geochemical pathways of groundwaters equilibration in granitic rocks. Based on the careful study of about 200 groundwaters collected from sites throughout Europe, we are now able to model the equilibration of such types of groundwaters (which at equilibrium are essentially alkaline and reducing) through a very small number of mineral/water reactions, temperature and concentration of mobile elements (mainly chlorine) being the two main parameters (Grimaud et al., 1990). Such a geochemical model allows us to successfully predict the distribution of a large number of major and minor elements. On the contrary, we are not able as yet to take into account the fate of trace elements. However, we have discovered that most III- and IV-valent trace elements are associated with the particulate matter and colloids: their behaviour is thus not simply described, in most cases, by a model of solid/solution equilibrium. We can thus expect that colloids, both inorganic and organic, may play an important rôle in the transport of a wide variety of radionuclides from a repository, including actinides (Moulin and Ouzounian, 1992). Hence, our programme involves an intensive study of the characteristics (nature, size, charge) and properties (notably the surface retention of heavy elements) of colloids either collected in natural sites or prepared in the laboratory. In recent years, we have made an extensive use of MeV ion beam techniques, such as the Rutherford Backscattering Spectrometry (RBS) for the investigation of the scavenging properties of such entities (see for instance Della Mea et al., 1992).

More recently, we have extended this approach to the study of the groundwaters equilibration in clay formations. One practical problem stemmed from the difficult collection of waters in such types of rocks, which are highly impervious and usually contain very small amounts of solutions. Under the coordination of ANDRA, and within the framework of a CEC project labelled ARCHIMEDE, we have been able to develop a specific device for water collection in the site of Mol, Belgium, and to propose a preliminary modelling of the pathway followed by such a groundwater towards equilibrium (Merceron et al., 1992).

All these hydrogeochemistry studies imply continuous analytical developments for the analysis of specific traces and ultra-traces as well as for in situ measurements (e.g. optodes devices; see for instance Motellier et al., 1993). We are in particular developing, in some cases in collaboration with other CEA teams, techniques such as capillar electrophoresis, laser spectrofluorimetry, polarography, etc.

Our interest also focuses, since a few years, on the metastable phases which tend to result from water/rock interactions (likely as well as from the alteration of various technological materials in the near-field). These phases are essentially amorphous hydrated (Fe)-Al-Si compounds, or 'gels', which have the extremely important property of efficiently 'trapping' a wide variety of heavy elements (Creach and Magonthier, 1992). Although not thermodynamically stable, the study of natural analogues suggests that these materials could remain amorphous for very long periods of time (hundreds of thousands or even millions of years). Our main interests deal with the retention capabilities of these compounds for a number of radionuclides or chemical homologues (e.g. lanthanides) and with their possible reorganization with time (e.g. crystallization) towards stable phases such as clay minerals, zeolites, etc.

An other important aspect of our research programme is the identification of retention mechanisms of radionuclides on mineral surfaces, as well as their modelling. Because, we think that partition coefficients (K_d), even if measured in carefully controlled experimental conditions, are of no scientific interest when applied to other conditions (e.g. natural ones), this mechanistic approach is necessary. In particular, we are currently proceeding to an intercomparison of the two models available for the description of such retention processes, namely the ion exchange model and the surface complexation model, the latter being presently the most widely admitted one. However, the former model is also supported by convincing experimental results obtained in our laboratory, notably for the description of elemental retention on multi-sites mineral surfaces (Ly, 1993). We thus think that the ion exchange model should be better considered by experts in this field. This is connected to the study of the elemental migration through rocks, which we are trying to take into account in our present models by coupling transport of fluids in porous media and simple chemical reactions such as ion exchange. In particular, the application of the code IMPACT, a chomatographic-like model developed by Sardin at ENSIC, Nancy (France), to rock columns made up of a mixture of sand and clay give remarkably predictive results (see for instance Fauré, 1993).

However, and whatever the importance of water/rock interactions and secondary phases formation processes, it is quite clear that the geochemical mobility of radionuclides in natural systems should strongly depend on their speciation. Therefore, we have carried out for years systematic studies on the basic thermochemistry of actinides (notably U, Pu, Am and Np) in order to decipher their speciation in various natural conditions of interest for the disposal of radioactive wastes. Notably, the carbonate system has been extensively investigated (see for instance Vitorge, 1992) and, more recently, we turned to the study of the chlorine and phosphate systems. This part of our programme involves both our participation to the OECD/NEA Thermodynamic Data Base (TDB) and specific laboratory experiments. Our aim in the TDB, is to contribute to the publication, as soon as possible, of the books of validated thermodynamic data on Np, Pu and Am, that on U having been recently published (OECD/NEA, 1992). The experiments carried out are designed to measure reliable values of complexation constants (for the abovementioned anion complexes) as well as standard oxidation potentials for these actinides. In particular, we have extensively studied the speciation of Pu in oxidative conditions and demonstrated the possible stability of Pu(V) in natural conditions (see for instance Capdevila et al., 1992). More recently, we turned to the investigation of the speciation and behaviour in natural systems of some important long-lived fission products (notably Tc and Se), from both the experimental viewpoint and by the study of the geochemical cycles of inactive homologues.

Throughout these studies, our methodological approach has always been to refer to natural analogues (as already mentioned above) in order to decipher 'how things are actually going on in Nature' (Petit, 1990; Petit, 1992a,b). During the last few years, several natural or archaeological analogues have been investigated (see for instance Petit, 1992b) for processes occurring both in the near-and far-fields. However, Oklo is probably the more crucial natural analogue which we have investigated, within the framework of an international project lead by CEA/IPSN and carried out under the auspices of the CEC. In effect, the newly discovered reactors (presently 16 are known), which have been preserved from any alteration subsequent to their functioning about 2 billion years ago, offer the opportunity to study a wide variety of processes likely to be important in a repository: stability of waste forms (including spent fuel-like UO_2 and bitumens), migration of radionuclides (including transuranic elements and lon-lived fission products) within reactors and in their immediate borders through the fissure system, long-range elemental transport by groundwaters, etc. Recent results on Oklo can be found, for instance, in Bros et al. (1993) and Hidaka et al. (1994). Moreover, a major project based on the design of modular mocks up to study the confinement of radionuclides, essentially in the near-field of a repository, is presently under definition.

Finally, during the last few years, we have also been involved in the development of a global strategy for the investigation of geological sites. In addition to general hydrology, structural geology, petrology, mineralogy, hydro- and atmo-geochemistry (both elemental and isotopic), thermo-chronometry, specific devices for the in situ collection of samples, in situ measurement of various geochemical parameters and in situ experiments have been developed (see for instance, for the AUTOLAB tool, Porcheron and d'Alessandro, 1992). This approach is (or will soon be) applied to natural analogue sites such as El Berrocal, a granitic formation located in the southwest of Madrid (Spain) and to sites where a remediation is necessary such as the direct environment of ancient uranium mines. In a few years, these tools could be mobilised, in the framework of the ANDRA underground laboratories, in association with the competences of other research institutions.

Materials science issues

Recent developments in this field intend to both ameliorate the confinement of radionuclides to match a goal of increased safety and to modify the current technological processes in order to notably decrease the amount of wastes to manage. The confinement can be increased, for instance, by using matrices demonstrating a better durability, including over very long periods of time, with respect to a variety of agressive parameters (radiation, groundwater, fire, etc.). These objectives could be of interest in the framework of a geological disposal of radioactive wastes but, since the law of 1991 (JO, 1992), their implication in the case of a long-term surface storage must also be thoroughly assessed.

Much basic research has been devoted to the development of specific formulations of cement-based materials for the embedding of a variety of low- and intermediate level wastes (Le Bescop et al., 1990). Difficulties are in particular raised when chemical interactions, possibly deleterious to the (long-term) behaviour of the matrix, can occur between the wastes and the material itself. This is for instance the case for the incorporation of boronrich ion exchange resins or sulphate-containings sludges. However, in all these cases, classical cements (e.g. OPC) for civil engineering were used. The same considerations would be true for containers, engineered barriers and various repository structures (where appropriate). Recent advances aim at developing cement-based materials with exceptional properties (so-called 'high- or very-high performance mortars and concretes) either through the use of special procedures for their preparation and/or of particular cements (e.g. aluminous cements). Three such properties are under investigation:

- the 'confinement capability' for major radionuclides of interest,
- the durability, in particular with respect to chemical agressions,
- the mechanical resistance.

The two first properties essentially depend on the porosity of the material which impacts the permeability, the ionic and/or molecular diffusivity, etc. The objective would thus be to achieve a porosity as low as reasonably possible. Such properties are investigated through the direct measurement of gas or mercury porosities and by the evaluation of diffusivities in specifically designed diffusion cells, notably for ^{3}H, ^{137}Cs and ^{22}Na. Some materials of this type have already been proposed by industrials and specialised laboratories, but they have to be tested for their performance in a 'nuclear environment' and, possibly, new ones developed. Recent measurements of ^{3}H diffusivities obtained on materials of the former category are quite promising in showing almost no transfer of this particularly mobile isotope over a period greater than 6 months. If entirely new materials are developed, with particular consideration for their environmental behaviour, the issue of their long-term durability and confinement capability should be particularly pinpointed, because no reference will be available. The research on these high-performance cement-based materials is still at a to early stage at CEA to be described in more details here. Rather, we have given some indications about the issues as we tend to consider them.

In recent years, particular efforts have been made at CEA for investigating the long-term resistance of cement-based materials towards irradiation (Bouniol, 1994) and aqueous corrosion (see for instance Revertégat et al., 1991; Adenot, 1992). Concerning this last item, Adenot (1992) has developed a model, where the transfer by diffusion of agressive ions and simple chemical reactions are coupled, which conveniently takes into account a variety of experimental results.. With the help of this model, we have been able to reliably predict the thickness of corrosion with time of cement materials used for surface disposal of low-level wastes. This model demonstrates that this thickness remains quite limited (a few centimetres, i.e. with no impact for the confinement performance) even for the longest period required by official regulations. However, such a model is presently limited to classical formulae of cements. Thus, it has first to be improved to correctly describe the corrosion behaviour of concretes and mortars (which are the actual materials used in industrial sites) and second, on the longer term, to be extended to entirely new materials such as high (or very high) performance materials.

In addition, the mechanical resistance, which in the field of radioactive waste management was not of particular importance until recently (compared to the two other properties addressed here), is now under a renewed consideration because is may be relevant in the case of a long-term storage of the wastes and/or of a reversible option for both storage and disposal. Here, we pay particular attention to the modelling of the mechanical properties because this approach is indispensable for the design of long-term storage concepts.

Readers interested in current developments for the improvement of other matrices should refer to Courtois et al. (1994) for bitumens and ceramics, used or considered for low- and intermediate-level wastes, as well as glasses, currently used for high-level wastes.

An other important area of R&D has to do with the isolation of wastes in the geological formation where the repository is emplaced. Because the action of groundwater is considered to be the main parameter for both the degradation of technological barriers (notably the matrix itself) and the transport of the radionuclides towards human beings, preventing its access to the wastes is a strategic goal for all concepts so far proposed. This goal can be achieved through two routes: i. in the classical concept, extensively studied by a large number of nations (including France) the wastes are protected from groundwater agression by a succession of barriers as impervious as possible ('Matriochka'-like concept initially investigated under the impulsion of the Swedish KBS projet in 1978). In particular, engineered materials essentially made up of bentonite, closely disposed around the waste containers, would play the rôle of a hydraulic barrier. Groundwater could thus flow in the host rock (though at a limited rate if the site is properly selected) but be derivated around the near-field of the repository by the presence of this insurmountable hydraulic barrier. ii. an other concept would stem from the consideration that, in particular in crystalline rocks, the water flow takes place in highly localised, discrete, fissures and fractures. If engineers were able to 'cut' the hydraulic network of active fissures/fractures at strategic points (with dam- or partition-like engineered systems), one could envisage to completely isolate a block of host rock from the general flow of groundwaters in that geological formation. This could possibly be done more easily and at a much lower cost than the (difficult) emplacement around hot and highly radioactive containers of classical compacted bentonite-based engineered barriers.

Indeed, after about ten years of extensive studies of the behaviour of classical engineered barriers (see for instance Beziat et al., 1992; Meunier et al., 1992;), we are now exploring this new concept under the auspices of ANDRA. We are currently investigating the thermo-hydro-mechanical properties of such dam-like systems, again based on a core of bentonite. These studies include laboratory experimentations, the build up of a 1/50th

mock up and modelling. We also intend, in a second step, to demonstrate in situ the feasibility of emplacement of such a system as well as its capability to efficiently interupt the water flow. It is too early to describe in details results which are still preliminary but significant data should be presented to the scientific and technical community within about a year.

VI. CONCLUSION

Because a law, voted at the end of 1991, has redefined the status of the R&D associated with the management of radioactive wastes in France, the investigations that we are carrying out at CEA in this field have been noticeably modified within the last two years. Studies have been launched on the advanced separation of long-lived radioisotopes and transmutation as well as on the improvement of radioactive waste matrices and long-term storage, all topics which have not been addressed here. In effect, we intended to focus the paper on the field in which we (that is to say the author and his team) have been most actively involved in recent years, namely the geological disposal of radioactive wastes, still a major option for the long-term management of transuranic and high-level wastes.

An important aspect of our studies in this framework has to do with the development of a methodology where qualification (including laboratory experiments and mocks up), in situ measurements, in situ experimentations and demonstrations, as well as natural analogues combine and complement each other. Its goal is to build up a model (or a series of local models) describing the (long-term) behaviour of a radioactive waste repository emplaced in a geological formation. We have illustrated our research programmes during the last few years both in the fields of geosciences and materials science by highlighting results of particular interests. Although much work will be needed when the French underground laboratories will be available and when the renewed concept for the confinement of radioactive wastes in geological formations will be specified, significant scientific and technological advances have already been made. In particular, they concern the basic chemistry of long-lived radionuclides, the geochemistry of groundwaters (including the speciation of trace elements), the properties of gels formed during water/rock interactions, the retention of species on mineral surfaces, the migration of radionuclides through porous rocks, and the in situ characterization of geological sites. In addition, results have been obtained on the confinement capability and the durability against irradiation and water attack of a variety of cement-based materials. High performance hydraulic binders are also under development and testing for application to both storage and disposal of radioactive wastes. Finally, extensive studies on a new concept of dam-like systems for the interruption of water flow in geological formations are also currently undertaken.

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RD&D for High Level Radioactive Waste Disposal in Sweden

Current status and future plans

Tönis Papp[1]

Background

The existing Swedish system for the management of radioactive waste has been developed systematically on the basis of proposals in mid-70s and on work initiated with the KBS Project during the latter half of the -70s.

The low and medium level waste from the reactor operations is stored locally at the reactor sites until transported to the final repository, SFR, situated close to the Forsmark nuclear power plant. A facility for interim storage of spent nuclear fuel, CLAB, is situated by the Oskarshamn nuclear power plant at Simpevarp. A seabased transport system for the transport of wastes is also in operation, see figure 1.

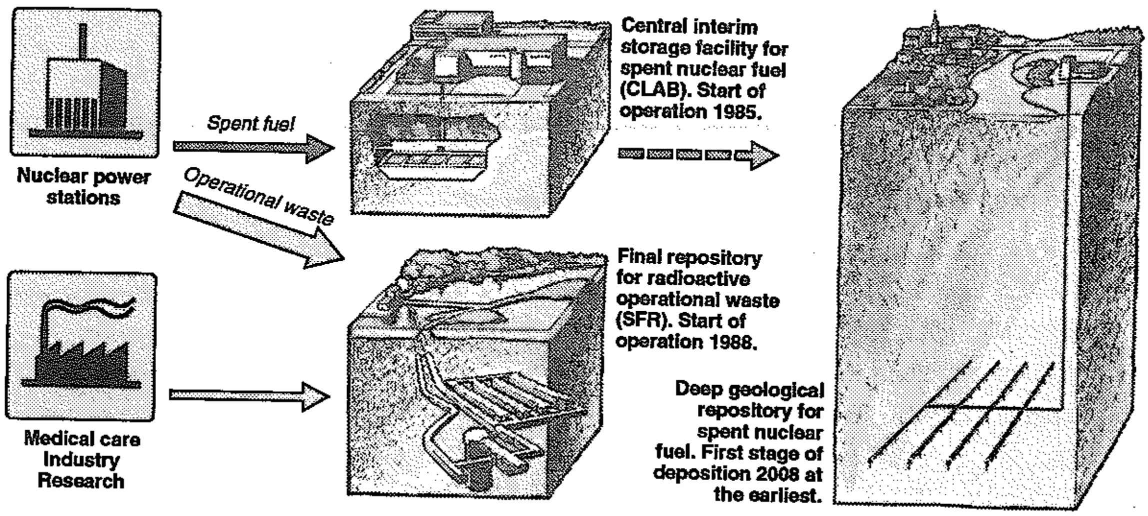

Figure 1: The Swedish radioactive waste management system

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Those parts of a complete system for management of all radioactive wastes in Sweden that have not yet been built are an encapsulation plant for spent nuclear fuel and a final repository for long lived wastes.

According to Swedish law, the Swedish nuclear utilities must together establish a comprehensive programme for the remaining research and development and other actions needed to take care of the radioactive wastes from nuclear plants in a safe way. Such a programme must be submitted to the Swedish Nuclear Power Inspectorate every third year.

On the behalf of the Swedish nuclear utilities the Swedish Nuclear Fuel and Waste Management Co – SKB – is responsible for all handling, transport and storage of the nuclear wastes outside of the nuclear power stations. SKB is also responsible to produce the required programme and to carry it through. The third programme, SKB RD&D-Programme 92 /1/, was presented in september 1992. It has been reviewed by the Swedish Nuclear Power Inspectorate and the National Council for Nuclear Waste. During 1993 the Government is expected to decide whether the program meets the legal requirements.

Beside the SKB RD&D programme the Swedish Nuclear Power Inspectorate and Swedish Radiation Protection Institute both have their own programmes to develop the competence and methodology needed for their licensing work.

General plan for further work

The work carried out during a period of about fifteen years in Sweden, and equivalent work in other countries, has led to a broad agreement among international experts that methods exist for implementing final disposal of high-level waste and spent nuclear fuel and that methods also exist for evaluating the long-term safety of such disposal. Expressions of this agreement include, for example, the approval of the KBS 3 report /2/ in Sweden and of similar studies abroad, and the "collective opinions" expressed by international expert groups in the OECD/NEA, the IAEA and the EC. An important conclusion in the most recent of these collective opinions /3/ is that further efforts should be focused on gathering and evaluation of data from proposed repository sites.

Proposals and alternative options for the final disposal of spent nuclear fuel have been reviewed and studied by both regulatory authorities and industry in extensive R&D projects during the 1980s. Thus, the important issues relating to encapsulation and final disposal of spent nuclear fuel in Swedish bedrock have been thoroughly elucidated.

Spent nuclear fuel contains large quantities of radioactive materials. Final disposal shall be arranged so that the waste is kept isolated in a safe manner while it has a higher radiotoxicity than otherwise found in nature, ie over a period of around 100 000 years. To bring about this isolation, a final repository for spent fuel is designed according to the multi-barrier principle. Safety assessments show that by using stable materials in the engineered barriers eg like in the KBS-3 concept, radioactive materials can be kept isolated for more than one million years.

The performance assessment SKB 91 /4/ shows that a primary safety role of the bedrock around such a repository is to provide a mechanically and chemically stable environment for the engineered barriers protecting the waste. Studies and investigations of the bedrock in Sweden during the past 15 years indicate that there are many sites possessing the properties and stability needed for constructing a safe repository.

After having examined safety, technical feasibility and other aspects for a number of different alternatives, work in Sweden has now reached a point where it should be concentrated to a main line. SKB has found that the knowledge today is sufficient for selecting a preferred system design, for designating candidate sites for siting a repository, for characterizing these sites and for adapting the repository to local conditions.

Thus the RD&D-Programme 1992 calls for completion of the research, development and demonstration work by building a final repository. This is to be done in stages, starting with a minor quantity around 10%. The main reason for the stagewise approach is the possibility to demonstrate the following :

- The siting process with all its technical, administrative and political decisions;
- The step-wise investigation and characterization of the repository site;
- The system design and construction;
- The encapsulation of spent nuclear fuel;
- The handling chain of spent nuclear fuel from CLAB to deposition in the repository;
- The operation of a deep repository;
- The licensing of handling, encapsulation and deep disposal, including the assessment of long-term safety:
- (retrievability of the waste packages);

Due to the time periods involved the post closure safety of the final repository cannot be demonstrated through field tests. The safety case must be based on a technical-scientific assessment of the performance of the repository over a long period of time.

When the first stage has been completed, the results will be evaluated before deciding whether or not to expand the facility to accommodate all the waste. This makes it also possible to consider whether the deposited waste should be retrieved for alternative treatment. The latter option means that it must be possible to retrieve deposited fuel during the period the facility is being operated for demonstration purposes.

The reason SKB is planning a stagewise development of a repository is not doubt as to the feasibility and safety of the disposal scheme. The plan should be viewed as an expression of an awareness of and respect for the fact that solutions of the nuclear waste problem arrived at by the R&D work need to be demonstrated concretely to people far beyond the circle of experts. It is SKB's opinion that a demonstration deposition of spent nuclear fuel with full freedom of choice for the future is a good way to enlist broad support for the method of disposing of the nuclear waste.

The final disposal of the spent nuclear fuel in Sweden is therefore planned to be carried out in two main phases: Demonstration deposition and final disposal. The work will extend over more than 60 years. The decision to take the step to final disposal will not be taken until after demonstration deposition has been completed, the results evaluated and other alternatives considered. These decisions lie beyond the year 2010. The plans in this programme cover activities needed to site and build the facilities for a demonstration deposition. It is SKB's judgement that the deep repository will later be expanded to full scale. However, it is not meaningful to discuss now the details of how this will be done.

Additional facilities and systems will be needed for a demonstration deposition of spent nuclear fuel:

- Encapsulation plant for spent nuclear fuel, including a buffer store for encapsulated fuel. The buffer store shall be able to be expanded so that it can be used for interim storage if the demonstration deposition is interrupted and the canisters are retrieved.
- Deep repository for encapsulated spent nuclear fuel.
- Transportation system between CLAB and the encapsulation plant for spent fuel and between the latter and the site of the deep repository.

Figure 2 shows a timeschedule for the facilities that are needed to dispose of the Swedish radioactive waste.

SKB believes that the first phase including the encapsulation plant and the deep repository up to the completion of demonstration deposition. can be completed within about 20 years. Thus, as is evident from Figure 2, it is possible to follow this up with final disposal of the remaining fuel and waste immediately after 2020 if the decision to do so is made in about 20 years.

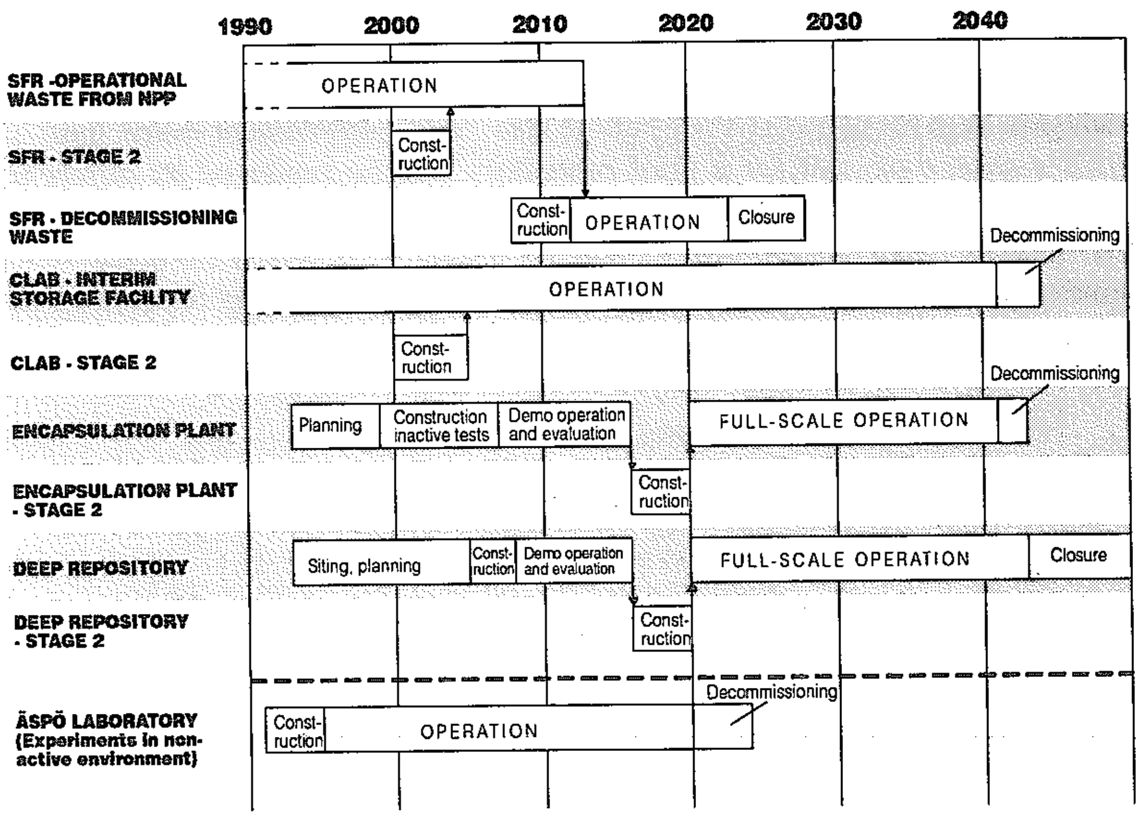

Figure 2: Preliminary time schedule for facilities needed for the final disposal of the Swedish radioactive waste

Siting

For the encapsulation of spent nuclear fuel, SKB plans to expand the central interim storage facility for spent fuel (CLAB). The spent fuel is already being stored at CLAB, and SKB believes that expansion of CLAB with an encapsulation plant for spent fuel has clear advantages in terms of logistics, resource utilization and environmental impact.

The siting of the deep repository is planned to take place in stages during the 90s. The selection of candidate sites for the repository will be based on the qualities necessary from safety-related, technical, societal and legal viewpoints. It must be demonstrated for the selected site and selected repository system that the safety requirements imposed by the authorities are met. It must be possible to build the repository and carry out deposition as intended. The siting process, the investigations and the construction work shall be carried out so that all relevant legal and planning-related requirements are met. And last, but not least, it shall be possible to carry out the project in harmony with the municipality and the local population.

Important for the planning of the siting process is the Government's decision regarding R&D-Programme 89: *"The Government notes that SKB's choice of sites for a final repository will be reviewed by different authorities in connection with SKB's application for permission to carry out detailed characterization of two such sites ..."*. Based on these guidelines, the work of siting and construction of the deep repository is planned to proceed in the stages seen in Figure 3.

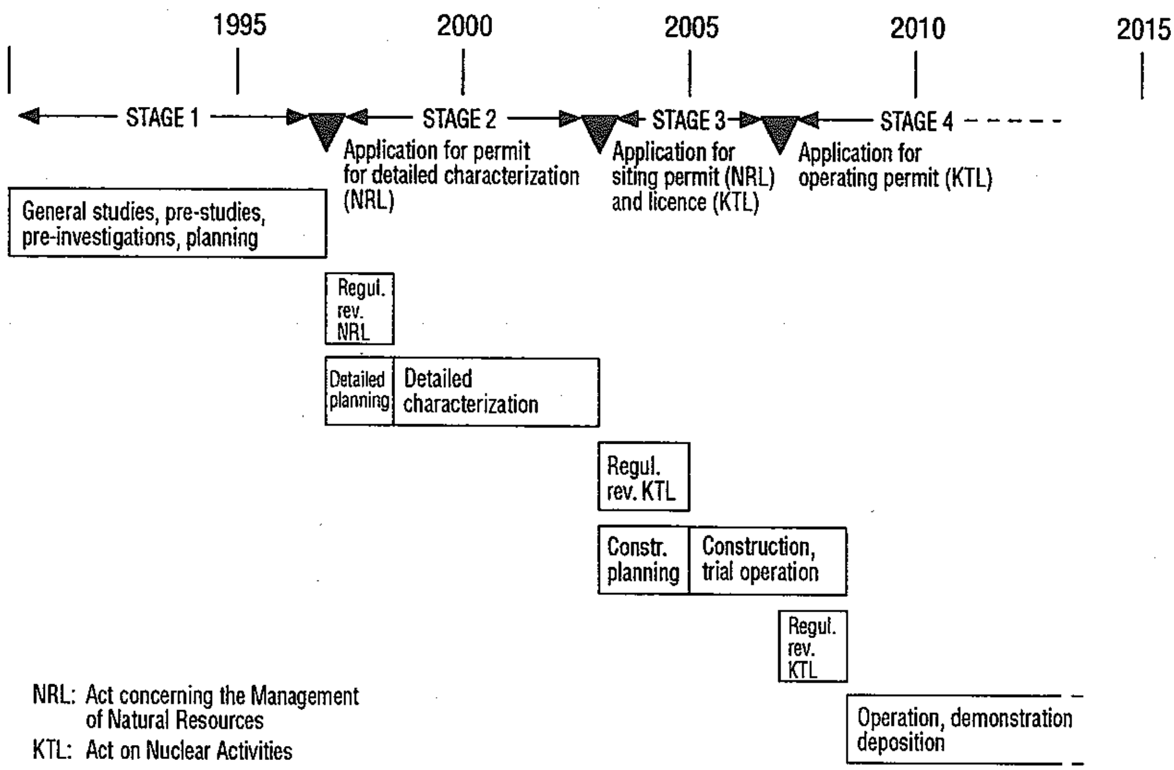

Figure 3: Preliminary timeschedule for the deep repository up to the completion of the demonstration deposition.

System for encapsulation and disposal of spent nuclear fuel

During the period 1986-92, SKB has studied alternative designs of a deep repository for disposal of spent fuel. Continued work on designing a deep repository for demonstration deposition should now be concentrated on one alternative to get the desired concentration and goal orientation in the future work.

Among canister designs studied, the copper/steel canister holding 12 BWR assemblies is chosen as the main alternative. This canister consists of a steel container, providing mechanical protection, surrounded by a copper container, providing long-lasting corrosion protection. Since the canister is a vital barrier, additional development will be conducted for the lead-filled copper canister as a reserve to the copper/steel canister.

Among repository designs, the KBS-3 design is kept as the main alternative.When adapting the repository to local conditions on the site, this design can be further optimized. At that time, related variants of the design can be given further consideration.

Outline of the R&D activities

For a demonstration deposition of spent nuclear fuel in a deep repository the encapsulation and deep disposal facilities must be developed. Encapsulation entails the final selection and testing of methods to fabricate, seal and inspect the quality of canisters, as well as to design, construct, license, install and operate the facility for encapsulation. Deep disposal entails siting, design, construction, licensing, installation, and testing of the repository and the deposition of the selected amount of spent fuel.

Safety assessments and supportive research and development are also required. Below the supportive R&D in some main areas are indicated.

Fuel studies

The fuel studies have the aim to progressively refine the models for radionuclide release. The studies of corrosion of spent fuel under different redox conditions, temperature and chemical environments will continue. The role of radiolysis on the redox conditions at fuel surface will be clarified.

Canister materials

The work aims at a selection of a copper grade by 1996 on the basis of creep deformation and creep failure as well as weldability. The studies of the premises for stress corrosion cracking will be concluded. The development of techniques for fabrication, welding and non-destructive testing will be supported.

Buffer and backfill

The work will be focused on completing a summary of the essential properties of various bentonites including cement/bentonite interaction. The design of a deep repository will be supported by studies regarding

- manufacturing techniques for the highly compacted bentonite around the canisters,
- grouting of fine fractures with cement or bentonite,
- techniques for backfilling of drifts and shafts with sand/bentonite.

Groundwater movements

Methods for describing the geometry of individual fractures and their hydraulic properties will be refined as will the interpretation methods for analysis and determination of hydraulic properties in the field.

The conceptual basis for numerical modelling will be examined with an emphasis on the dependence of the flow pattern on rock stress, fracture mineralization and permafrost depths. The effect of brief pressure change in the groundwater due to earthquakes will be investigated.

The hydrogeological premises for groundwater flow and transport in a regional perspective will be further explored. Consideration will be given to today's climatic situation, future glaciations and deglaciations.

Stability of the bedrock

A summary will be compiled of the principal load directions that have impacted the Baltic Shield during its historical brittle-tectonic period. The directions of dykes, previous sediment indications, erosion traces, fracture mineralization etc. will be investigated in a regional perspective. Recent plate-tectonic processes and glaciations will provide background material for such a summary.

Recent studies of the risk for earthquakes in the Nordic countries will be compiled, discussed and evaluated jointly by TVO and SKB. The study of methods for dating recent movements in fracture zones will continue.

Geohydrological and rock-mechanical calculation models

A study will be made of how the volume representativeness and dimensionality of hydraulic data can be incorporated in a model structure. Site-specific stochastic groundwater modelling with indicator simulation shall also be able to integrate and take into account general geological and geophysical information in the conductivity distribution.

A regional flow model is developed for conditions during glaciation and deglaciation based on a scenario for future climatic conditions.

Groundwater and fracture mineral chemistry

The chemical interaction between rock and groundwater and the influence of mixing processes is important. Sampling and equilibrium modelling of near-stagnant groundwater from tunnels in Äspö is planned. Distribution of trace metals between groundwater and fracturefilling minerals will be studied as an analogue for radionuclide retention.

The importance of acidification, redox reactions and microbial processes is being explored.

Radionuclide chemistry

Thermodynamic data for solubility and speciation of actinides in deep groundwaters will be determined. Co-precipitation and its importance for nuclide migration are being studied.

The groundwater's content of organic complexing agents, colloids and microbes is investigated at Äspö and in the framework of international analogue studies. Laboratory experiments are currently in progress.

Sorption effects in nuclide transport are today modelled with K_d. Development of surface complexation models is in progress and may eventually replace the K_d concept.

Validation of the processes in transport models and nuclide migration

Weakly sorbing tracers for migration tests will be developed and tested in the lab for later use in field tests at Äspö. Groundwater flow is being validated with tracer tests with non-sorbing tracers. Further experiments of this kind will be carried out at Äspö. The natural tracers already present in the groundwater are being used for large-scale studies of groundwater flow.

The development of a special chemical probe, CHEMLAB, for in-situ migration tests in boreholes will be concluded.

Biosphere studies

The investigations aim at a better quantification of the uncertainties that stem from the fact that the biosphere is constantly changing. The database for biosphere dispersal models will be further developed and analogous dispersal processes in nature will be used for validation.

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Current Status of R&D Programme for Deep Geological Disposal in the United Kingdom.

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1. INTRODUCTION

This paper gives an overview of the situation regarding the disposal of radioactive waste in the UK. It also describes some of the current R&D activities, relating to the preparation of the safety case for a proposed deep underground repository. In particular, the paper concentrates on the research activities being undertaken for UK Nirex Ltd, the organisation responsible for the disposal of low- and intermediate-level radioactive waste (LLW and ILW respectively). Other research, not covered in this report, is being carried out on behalf of the regulatory Departments of the UK Government. These Departments have the responsibility of assessing the safety case that will be presented by Nirex as part of the licensing procedure.

2. OVERVIEW OF RADIOACTIVE WASTE DISPOSAL SITUATION IN THE UK

2.1 Background

ILW and high level waste (HLW) are currently stored at nuclear establishments. As part of the Government's strategy for the disposal of ILW, Nirex has the task of developing a deep underground repository for the waste.

LLW constitutes the majority by volume of all radioactive waste. The waste arises not only from the nuclear industry, but also from all users of radioactive substances, such as hospitals, research establishments and industry. At present, it is mainly disposed of at a 300-acre controlled landfill site at Drigg in Cumbria, operated by British Nuclear Fuels plc. It is packed into containers and placed in concrete-lined trenches which are subsequently sealed. Original plans for the long-term disposal route for LLW involved disposal in a new near-surface repository. However, in 1987, this proposal was abandoned, and the Government agreed that the deep repository should also be used for some LLW. Current plans also include the continuing use of Drigg.

Following extensive, systematic site selection exercises, Nirex is currently undertaking detailed geological studies at Sellafield, also in Cumbria, as a potential site for the deep repository. The potential repository host rock is a tuff unit of the Borrowdale Volcanic Group of rocks, overlain by approximately 400 m of Permo-Triassic sediments at the location of interest. Current plans are for the disposal of approximately 300,000 m^3 of ILW up to the year 2030. The LLW volumes estimated to arise to this time are approximately 700,000 m^3 (taking account of supercompaction)[1]. The design for the repository will be based on a multi-barrier concept; ILW will be encapsulated by grout within steel and concrete containers which, on emplacement in the repository, will be surrounded by a cementitious backfill.

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2.2 Regulatory Requirements

With regard to the long-term ('post-closure') safety of the proposed deep repository, the regulators have published principles that will form the basis of any authorisation for disposal of wastes[2]. The principles include the following requirement, that 'the appropriate target applicable to a single repository at any time is a risk to an individual in a year equivalent to that associated with a doses of 0.1 mSv, about one chance in a million' (of early death from radiation exposure).

2.3 Performance Assessment Methodology

In considering the risk posed by the repository, the first step is to identify the pathways by which radionuclides may be returned to the environment[3]. There is an international consensus on the need to address the following four pathways:

- transport in flowing groundwater;
- release of radioactive gases;
- human intrusion into the repository or the radioactive plume in the geosphere;
- natural disruptive events.

A deep repository is well isolated from extreme events on the surface. The only natural event identified that might potentially transfer bulk quantities of radioactive wastes to the accessible environment is the impact of a large meteorite. The probability of this occurring is very small, and the associated risk is judged to be negligible.

For each pathway, the controlling processes have to be identified and quantified, then mathematical models are constructed to describe them. Once the models have been developed, appropriate data are required. These pathways are not completely independent. Pathway interactions also have to be considered. An example is the potential alteration of the groundwater pathway by gas overpressurisation and subsequent rock fracturing. A programme of scenario development and probability assignment is in progress to arrive at a more thorough description of repository evolution.

In the next section, research associated with groundwater and gas pathways is described.

3. OVERVIEW OF R&D PROGRAMME FOR DEEP GEOLOGICAL DISPOSAL

To progress its plans for waste disposal, Nirex has been supporting a research & development programme since 1982. The 'Nirex Safety Assessment Research Programme' (NSARP) forms the main R&D effort[4]. In addition there are extensive site investigation activities in progress, including a deep borehole drilling and core characterisation programme. The main objectives of the borehole programme are to study control upon water flows in the rocks surrounding the proposed repository location, and the hydrochemical and mineralogical nature of these rocks. A further planned phase of the site investigation programme is to develop an underground research laboratory, the Rock Characterisation Facility (RCF)[5]. The NSARP will be developed to guide the design of experiments to be carried out in that facility at the appropriate time. The research programme and site investigation activities are closely linked with the performance assessment studies, which are carried out by the Disposal Safety Assessment Team (DSAT). In the rest of this section, an overview of the NSARP is presented.

3.1 Nirex Safety Assessment Research Programme (NSARP)

The NSARP has a number of aims, which are broadly summarised as follows[6 7 8 9 10]:

1) to understand the important physical, chemical and biological processes that govern the transport of radionuclides away from the repository;

2) to construct models that provide a good representation of these processes, allowing the description of what is currently observed and the prediction of future evolution;

3) to provide data, by laboratory and field investigations for the important parameters and radionuclides;

4) to provide advice to Nirex on matters such as repository materials;

5) to liaise with the DSAT, so that research is focused on key processes and parameters.

The near field of the repository is taken to encompass the repository and its contents. The analysis of the near field is concerned with the physical and chemical containment of the radionuclides and defines the flux of radionuclides leached from the repository into the surrounding rock. The far field is the term used to describe the geology surrounding the repository. The assessment considers the transport of dissolved radionuclides through the rocks, and considers chemical and physical retardation processes. The biosphere is the term used to describe the human environment, and in the performance assessment, the analysis is concerned with the transfer of radionuclides from the geosphere to the biosphere, and uptake by humans.

3.2 Groundwater Pathway

In the context of a deep repository, the major barriers to the return of the radionuclides to the biosphere are:

- physical containment within the wasteform and repository;
- chemical containment within the repository;
- a long groundwater return time (to the biosphere) in the geosphere;
- retardation of many of key radionuclides in the geosphere relative to the groundwater flow.

<u>Prediction of physical containment of the engineered barriers</u>

Studies are in progress to investigate the containment ability of carbon and stainless steel waste containers under repository conditions. The main threat to canister integrity is corrosion. Under high pH conditions within a cementitious backfill, rates of corrosion are generally very slow. However, under aerobic conditions, predicted to exist for the first few hundred years, faster forms of corrosion are possible[11]. The research programme involves both experiments and supporting mathematical modelling. The models are based on a mechanistic understanding of the corrosion processes, and provide a predictive capability. Results from these studies are used in detailed assessments of canister containment behaviour, which consider both the design of the canister, and the different modes of corrosion that operate through its lifetime.

Experimental and mathematical modelling studies are also in progress into the effects of cracks in the cement on the release of dissolved radionuclides. These cracks could become preferential pathways for groundwater flow.

The potential enhancement of radionuclide mobility by colloids has received attention in the NSARP. The research suggests that a relatively small number of colloids are generated in the near field from the backfill. This is thought to be associated with the low silica content of the current formulation of the backfill material.

Prediction of chemical environment in repository and its evolution with time.

Detailed research is being undertaken to predict the way in which the repository will evolve over long periods of time. In particular, changes in pH in the repository porewater through progressive leaching of the soluble components of the cement backfill, the evolution of the oxidation conditions under the influence of the corrosion of metal canisters, the evolution of soluble products of the degradation of organic materials in the waste, the evolution of temperature in the repository and the formation of colloidal material are all important areas of research. The most important parameters for the performance assessment that are affected by such chemical changes are the solubility and sorption of radioelements. Key results in these areas are summarised below.

Studies have been performed to investigate the long-term chemical control of the pore water by the cement. Present estimates suggest that the porewater will maintain a pH of over 10.5 for a period in excess of 10^5 years[7]. Research is also in progress to investigate long-term changes to cement minerals with regard to both the pH buffering capacity and changes to sorptive properties. Mathematical modelling of the oxidation potential of the porewater in the near field, using the CHEQMATE code[12], has shown that the maximum time for reducing conditions to be established is a few hundred years[7].

Solubilities of the key radioelements have been measured under a range of possible conditions within the repository[13,14]. This work has been underpinned by mathematical modelling studies, using the HARPHRQ program[15] and HATCHES database[16]; these studies help provide an understanding of the key processes. Also, sorption coefficients for key radioelements onto cements have been measured and for the actinides, in particular, sorption was found to be high (>1 m^3 kg^{-1})[17]. In order to increase the understanding of this process, experiments involving sorption onto individual cements phases have been carried out; no particular phase dominates the overall process[18].

A large body of work to investigate the effect of the degradation of organic materials (by chemical, radiolytic and microbial activity) has been performed[19 20 21]. For the UK waste streams, cellulose has been identified as important in terms of the potential impact on radioelement solubility and sorption behaviour. At high organic loadings, solubilities can increase by many orders of magnitude and sorption can also be reduced significantly[22]. However, at the loadings anticipated within the repository, the effects are much smaller. Although chemical degradation of cellulose under alkaline anaerobic conditions generates several products, 2-C-(hydroxymethyl)-3-deoxy-pentonic acid (iso-saccharinic acid, ISA) has been identified as a key complexant for plutonium and other radioelements. Solubility measurements in solutions of ISA and similar compounds are supported by interpretative mathematical modelling[23]. Studies on cellulose are continuing in order to provide further data on degradation processes, and on the effects of the products on radioelement behaviour to confirm that the effects within the repository will be limited.

The solubility studies have been supported by natural analogue programmes. The most recent is the study of the interaction between a hyperalkaline groundwater and a sedimentary rock in Jordan[24]. Geochemical modelling programs have been used to predict the concentrations of key radioelements under these conditions and these predictions have been compared with the concentrations in the groundwater measured at the site. Studies have also been carried out of the mineralogy in the vicinity of the groundwater and the association of trace elements. Changes to the mineralogy of the rock surrounding the high pH source are also being studied.

Groundwater flow

Groundwater flow is largely driven by topographically-determined pressure heads and water fluxes are determined by the permeability of the various geological strata and features. There are two different mathematical approaches to describe groundwater flow in rocks: a continuum approach and a fracture network approach. Continuum modelling is appropriate for rocks in which the dimensions of the pores are much smaller than the lengths of interest, such as clays. It is also appropriate for modelling the overall flow on appropriately large scales in rocks containing fractures that are much larger than pores in clays. More detailed study of the flow on the scale of the fractures can be undertaken using the alternative fracture network modelling approach.

A number of groundwater flow programs are used for the Nirex performance assessment studies. These include the NAMMU[25] program, which is uses a continuum model approach, and the NAPSAC[26] program, which simulates flow and transport in fracture networks. One of the key issues that has to be addressed is the heterogeneous nature of rocks, where properties can vary by orders of magnitude. Various methods of dealing with spatial variability in modelling have been considered and test cases carried out as part of the INTRAVAL project[27].

In hard volcanic rocks, such as the Borrowdale Volcanic Group at Sellafield, groundwater flow takes place predominantly in an interconnected network of fractures. In order to set up an adequately defined model of such a system, it is necessary to have information on such fracture properties as their effective hydraulic apertures, spacings, lengths and orientations. In practice, detailed information on these properties can only be obtained for a sample of the fractures. A stochastic approach is therefore adopted, in which the distributions of fracture properties are determined experimentally. A model using NAPSAC has been set up based on data on individual fractures obtained from a field site in Cornwall, where the fractures are at shallow depth (10-30m) and have an average separation of 20 cm. The results from the model have been compared with measured effective permeabilities inferred from a series of flow experiments performed over representative lengths of boreholes containing many fractures. These agreed to within factors of two or three[28].

Laboratory-based experimental programmes are also used to measure physical properties of rocks that are important input data for the mathematical modelling. These properties include diffusibility, porosities and matrix permeabilities[29].

Transport of radionuclides in the geosphere and retardation

For non-sorbing radioelements, rock-matrix diffusion is a potentially important retardation process in a fractured geosphere, as at the Sellafield site. For sorbing radionuclides, this process gives access to additional sorption sites away from the fracture. Experimental techniques are being developed to quantify the rock-matrix diffusion properties of rocks from Sellafield, and to understand the influence of effects such as channelling. Experiments to investigate flow and transport involve blocks of rock containing single fractures; the largest used in experiments to date had dimensions 0.6m by 1m[30]. The technique of Positron Emission Tomography (PET) has also been applied to the study of mass transfer in single fractures, and it has proved possible to visualise channelling of flow and aperture variation within the fracture plane[31]. These data are used to improve confidence in the use of mathematical models that describe flow and transport in single fractures.

Sorption of the radionuclides onto rock surfaces is one of the major retardation processes in the geosphere. A large laboratory programme is being undertaken to measure distribution coefficients of key radioelements onto the geological materials[22]. Experiments include batch sorption, coupled diffusion-sorption techniques and application of surface analytical techniques to improve the understanding of the sorption mechanisms, and aid justification of choice of sorption parameters for the performance assessment. These studies are again underpinned by interpretative mathematical modelling. The sorption studies also address the effects of organic degradation products from the near field on sorption onto rocks and the potential alteration of the host rock by high pH porewaters emanating from the repository.

The laboratory programme is supported by field studies of natural geochemical systems[32]. Mineralogical distribution of uranium in the rocks are characterised and the importance of iron oxide minerals in retarding uranium migration in natural systems has been identified. The studies have involved the development of sequential extraction schemes to allow identification of the chemical locations of uranium and thorium in rocks. Uranium-series disequilibrium measurement are used to estimate in-situ retardation factors, and these are used to improve confidence in the use of laboratory data in performance assessment modelling.

Studies are in progress to investigate the role of colloids in the geosphere in the enhancement of radionuclide transport. In particular, the work aims to characterise the colloids, investigate their mobility and quantify their capacity for radionuclide sorption. Field sampling of natural colloids has been carried out at three sites, including a fractured slate site at Cornwall[33]. A range of colloid types were identified, including silica, alumino silicates, iron oxides and organic colloids, but in all cases, only a small fraction (<10%) of the uranium and thorium in the groundwaters was associated with the colloids. Further work involves laboratory and field experiments to assess the mobility of colloids in rocks. Synthetic monodisperse silica and iron oxide colloids have been prepared and their transport through columns of glacial sand has been investigated[34]. Results have shown that the iron oxide colloids are less mobile than the silica colloids. Studies relating to colloids are continuing.

Biosphere

This part of the research programme comprises an integrated set of projects relating to climatology, geomorphology, near-surface hydrology and radionuclide transport in soils, as well as various review studies on the behaviour of specific elements in the environment and of particular environmental processes of potential significance[10]. Much of the work in this area is undertaken by universities in the UK.

On the long timescales of relevance to the assessment of post-closure safety, climate change[35] is of particular importance in defining the extent to which the land surface in the vicinity of the repository will be modified. It also defines the range of ecosystems that may occur at different times in the future. Studies of climate are complemented by geomorphological investigations, which provide information on important processes and rates of landform evolution.

In turn both these studies provide a basis for models that are used to assess the near-surface hydrology in the future; this is important for understanding the distribution and transport of radionuclides, in the environment following their release from the geosphere[36]. Such work is complemented by detailed experimental and modelling work examining radionuclide transport in soils and uptake by crops[10,37].

3.3 Gas Pathway

A substantial volume of gas will be generated in the repository. Hydrogen will be generated by corrosion of metals in the anaerobic environment that will arise some time after repository closure and the degradation of organic wastes by microbial action will yield carbon dioxide and methane. The radionuclides H-3 and C-14 may be incorporated into these gases and be transported to the surface relatively rapidly in

conducting fractures in the rock. Potential concerns include the radiological impact and the potential build-up of pressure in the repository and surrounding geology.
A model of gas production and migration (GAMMON)[38 39] has been developed. Input data are provided by experimental programmes on the corrosion of metals[40] and microbial activity. Models used to study the migration of gas through the near field and geosphere include the PORES program[41]. Validation studies are being conducted using data from field tests[42 43].

4. ACKNOWLEDGEMENT

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HLW disposal in Switzerland: Current status and future R&D focus

C. McCombie[1)]

Background

Switzerland is a small, mountainous country with a high population density. With a well developed economy, secure energy supply is of considerable importance but, as it is poor in natural resources, it must import about 80 % of its primary energy needs (~ 65% oil products and 10% natural gas). Electricity covers about 20 % of energy demand, of which about 40 % is supplied from nuclear plants with most of the rest being hydroelectricity.

The main source of Swiss radioactive wastes is nuclear power production. First nuclear power production was in 1969 and Switzerland currently has 5 nuclear power plants (pressurised water reactors and boiling water reactors) with a total capacity of almost 3 GW(e). To date, most Swiss disposal planning focused on waste returned from foreign reprocessing plants, but currently the preferred strategy of the utilities is to keep both options (reprocessing or direct disposal) open for the 2/3 of the total spent fuel inventory for which no reprocessing committment has been made. Additionally, it is currently assumed that there will be no expansion of the Swiss nuclear programme and that the current reactors will not be replaced at the end of their operational life (40 years). This scenario leads total spent fuel arisings of ~ 3000 tU. If all this waste were reprocessed Switzerland would receive only very low total volumes of HLW - about 500 m^3 which is returned as ~3000 canisters each containing 150 l of vitrified waste. It is planned to store vitrified HLW or spent fuel for at least 40 years prior to disposal and specific projects are underway to provide the required intermediate storage facilities.

Since storage times could readily be extended even further, there is no urgent technical requirement for a national HLW disposal facility. Indeed, given the small quantity of HLW, it can be argued that disposal as part of a joint (international) project would be more attractive on economic and technical grounds and this option is expressly left open at present. Nevertheless, an extensive, national HLW programme is

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underway with the aim of demonstrating that, if required, a repository providing the required levels of long-term safety could be constructed at at least one site within Switzerland.

In the current phase of the HLW programme, most effort is devoted to the question of demonstrating siting feasibility and the strategy to reach this goal has recently been defined for the period up to the year 2000.

Repository siting

A 3-phase siting strategy was conceived at the start of the eighties. In a first Phase I, regional studies are based on widespread borehole data as well as extensive observations and measurements from the surface. This leads to a more localised Phase II in which intensive investigations (e.g. closer boreholes, 3D seismics) explore in more detail the siting potential of smaller areas. A subsequent Phase III involves the major tasks of shaft sinking and exploration at depth, leading to full characterisation of a given site. Selection criteria for Phase I are purely geological and hydrogeological; already at Phase II, however, project work is sufficiently localised that planning issues and socio-political aspects can be considered in more detail and communication and dialogue with potentially affected communities becomes of major importance.

Original desk studies at the end of the seventies covered all areas of Switzerland and a very wide range of host rocks. Following the studies, the crystalline basement of Northern Switzerland was allocated top priority and a Phase I characteristation programme was initiated. The regional crystalline characterisation programme has been a major, 10-year exercise. In the course of investigating a 1200 km^2 region, 7 deep boreholes (1300 - 2500 m) were drilled, 700 km of seismic measurements performed and comprehensive hydrochemical, hydrological and neotectonic studies carried out. At present all of the field results are being summarised and analysed in a regional geological synthesis with an associated total system assessment (Kristallin-I).

Already in early work around 1980, the sedimentary formation, Opalinus clay (OPA), which lies under much of the Northern part of Switzerland in a relatively homogeneous, low-permeability layer of rather restricted thickness (80-120 m) was identified as a promising host rock option. Nevertheless, intensive sediment studies which were initiated in 1986/87 evaluated 7 potential formations which were narrowed in to two candidates Opalinus clay (OPA) and Lower freshwater Molasse (USM).

Nagra assigned OPA first priority for regional characterisation-based, predominantly, on the relative simplicity of the structure of this formation. For the OPA option, a new field campaign involving some 250 km of seismic measurements complementing existing data was carried out during the winter of 1991/92. For the USM, extensive existing seismic data were gathered during exploration for hydrocarbons and these have been critically evaluated.

In the near future, key milestones are the completion of the Kristallin-I analysis and current stage of the OPA Phase I programme. For both the crystalline and the sediment project, specific proposals have been developed for a technical programme for the next stage of field work and suitable areas have been identified. Rather than concentrate solely on one option for the siting feasibility project, it has been decided to spread the available resources over both projects. This approach is intended to increase the acceptability of the siting feasibility project to be presented to the Government around 2000.

The work to date in OPA allows identification of several local areas which can be investigated in detail. One of these will be chosen and a siting feasibility project based on data from a further deep bore-hole and from 3-D seismic measurements will be prepared by ~ 2000. If technical suitability and safety of OPA for a deep repository can be convincingly demonstrated for the chosen site, other sites in the same formation are expected to be also suitable.

For crystalline rock, a feasibility project is expected to be much more site-specific. At a promising site identified during the Phase I regional field programme, detailed seismic investigations and also deep boreholes (deviated in order to intersect sub-vertical faults) will be carried out. The Phase-I crystalline syntheses, currently being documented, indicates that the suitability requirements on the crystalline host rock are relatively modest, so that the likelyhood of demonstrating siting suitability at the chosen test site is reasonably high, although negative results can obviously not be excluded.

Fundamentals of the HLW repository and safety case

The current conceptual repository design was developed taking into account the potential geological host rocks, the very low volumes of HLW expected and the very severe government requirement of demonstration of safety of waste disposal as a condition of extending reactor operating licenses. The concept, illustrated in Figs. 1 & 2 for the crystalline host rock option, has the following features:

1) extremely deep disposal (up to 1 km below surface) in a purpose-constructed facility

2) in-tunnel emplacement of waste packages

3) Very massive engineered barriers; in addition to the vitrified waste in its steel fabrication canister, a 25 cm thick steel overpack is envisaged which is surrounded by a >1 m thick anulus of highly compacted, purified bentonite clay. The 150 l glass block containing ~ 30 kg of radionuclides is thus surrounded by ~8 tonnes of steel and 50 m^3 (85 tonnes, dry) of bentonite.

The performance of this repository system was analysed in the Project Gewähr 1985 study for the crystalline host rock (Nagra, 1985), in the Sediment Intermediate Report for both OPA and USM (Nagra, 1988) and an updated crystalline study is in the final stages of documentation (Kristallin-I; Nagra 1993). For each of these studies, the safety case is clearly dominated by the high performance of the engineered barriers. Indeed, even for cases in which the geosphere barrier is short circuited and near-field releases considered to pass directly into the biosphere, the maximum calculated dose is not significantly higher than in the base case and is well below regulatory guidelines.

Near-field performance is very "robust" (McCombie et al, 1991) in that it does not take credit for all possible processes which could decrease releases and is relatively insensitive to variation of uncertain parameters within reasonable ranges (McKinley et al, 1992). As long as the geology provides a suitable environment for ensuring engineered barrier longevity, the calculated performance varies little between the potential host rocks considered. The parameters used in the near-field analysis and the results of the modelling have been compared with those from other PAs (Neall, 1993). In particular, because of close similarity of concepts, a detailed comparison with PNC-H3 has been carried out. The Swiss conclusions with regard to near-field robustness due to the hydraulic/chemical buffering rôle of the bentonite, low radionuclide solubilities and low glass degradation rates are seen to be consistent with other recently published studies.

Characterisation of the localised fissure systems carrying advective water flow which occur with low frequency in the rocks of interest is especially difficult at the present stage of investigation, when sampling is limited to a few deep boreholes. Thus, although the characteristics of the geological barrier may be so good as to ensure that effectively no releases occur over the timescale for which quantitative analysis is meaningful (~ 1 million years), uncertainties about possible "fast flow" paths mean that a much more pessimistic representation of the geosphere must be used for robustness.

Future R&D priorities

Some important R&D objectives for the future are aimed at being able to take more credit for the performance of the geosphere by improving our understanding of the hydrogeological and radionuclide transport properties.

As just mentioned, a main problem consists in identifying and characterising the major water-conducting faults. Important examples of research needs arising in this area are the further development of combined characterisation methods, based on structural geology, hydrologic testing, and geophysics, and of systematic exploration strategies aimed at identifying the minimum necessary field data. A further key issue in hydrogeological modelling concerns the quality of predictions of the dilution potential for groundwater emerging from low permeability geologic formations into nearer surface strata or into the surface hydrological regime. Dilution of this type can contribute large factors of safety for releases at far future times and it is thus important to produce sufficiently reliable estimates.

For near-field hydrology much more information will be available in the future when a shaft and test drifts have been excavated. In addition to quantifying the geometry and hydraulic properties of natural fracture zones around galleries and shafts, a major problem is quantifying the influence of the damaged-zones around excavations. These will intersect transmissive elements, and such a zone could induce shortcuts and render sealing less effective. Up to now, attempts at direct measurements of excavation-damaged-zones on the near-field hydrology effects have been largely unsuccessful. Further specific issues involving water flow around tunnels are the effects of drying-out and resaturation of the rock. The former is important because unsaturated zones around open tunnels may affect results of, and conclusions drawn from, in-situ experiments; the later resaturation of a filled repository may be an important long-term transient effect influencing performance during the first phase.

Radionuclide transport is determined by geometrical factors, the network of water conducting features along a pathway for groundwater flow, and by the mechanisms involved in transport. Pathways must be characterised by data from bore cores, bores holes and drifts, and then transformed into a conceptual model. The data density will never be sufficient for a detailed modelling and will give little information on the interconnectivity of the various flowpaths. However, natural analogues help to develop model concepts and many processes and their interactions can be studied by field experiments in a quantitative way. When carefully planned and performed (and complemented with well-chosen laboratory back-up investigations), such field experiments help to increase our understanding in tracer transport and our confidence in predictive modelling. The migration experiment at the Grimsel Test Site is a good example of cooperative Swiss/Japanese work which is making valuable contribution in this area. Largely unresolved is still the problem of scaling up from field to repository conditions in space and time. Very often the question is less whether a particular process, e.g., matrix diffusion, is operative, but rather its magnitude and its effectiveness over the long time spans that must be considered. Here again, observations on natural analogues can provide valuable understanding and even bounding data.

These key geotechnical, hydrological and geochemical issues are being tackled also within the scope of the Nagra research programme in the Grimsel underground laboratory for which a further 3 year operational phase is being planned.

In the performance assessment area, R&D can be usefully focused in 4 areas in the near future; extension of analyses to consider spent fuel and TRU, development of more realistic near-field models in order to compare potential host rocks and optimise designs, improving understanding of key processes and of corresponding databases and development of approaches to convince regulators and the general public of the acceptability of the concepts developed. The studies on TRU waste in particular, present a great challenge due to its relative heterogenity, chemical complexity and large volume (which precludes use of massive barriers of the types used for HLW). Although the inventory of radionuclides in this waste is orders of magnitude below that in HLW, release rates from the near-field may be higher and retardation in the far-field may be difficult to assess due to complications caused by associated hyperalkaline leachate.

In order to compare potential host rocks and to specify requirements in geology, development of near-field models which are, at the same time, as realistic and robust as possible is necessary. Thus, effort is required to provide further information on potential perturbations which, although unlikely, may affect performance - e.g. microbial activity, colloid transport through bentonite, gas evolution and release, bentonite erosion. Although probably extremely conservative, selection of elemental solubility limits is a fairly subjective process and further effort to develop more realistic mechanistic models (e.g. including co-precipitation) would be valuable. Over-conservatism in the model would also be decreased if credit was taken for transport resistance in the failed canister/leaching glass block and, in particular, for nuclide uptake (sorption and/or co-precipitation) in canister corrosion products.

For many elements, sorption is an important retardation mechanism in both the near-field and the far-field. Within performance assessment, sorption is taken into account in the transport model by a retardation factor or a retardation function. Experience over the last few years has shown that sorption distribution ratios extracted from well designed dynamic migration experiments and static batch sorption experiments are consistent: the difference occasionally claimed can generally be traced back to an inadequate (e.g., single porosity, linear sorption) transport model. Although a sorption data base adequate for performance assessment purposes can be compiled from empirical laboratory measurements, it would be preferable to have a better mechanistic understanding of the sorption processes. Considerable progress has been made in the last decade in developing surface complexation and ion exchange models. Workers in waste management could take more advantage of the research done in other fields and apply such concepts to their systems. The sorption data bases for transport modelling are still very rudimentary, but phenomenological isotherms could be measured for more of the important radioelements and this would certainly improve model realism.

Finally the development of better approaches to present the results of assessments is required. For example, the gradual increase in uncertainty in parameter values with time could be shown by various graphic techniques and, especially for the general public, the significance (or lack of it) of very low doses could be illustrated by comparisons with natural radiation sources. Natural analogues can be particularly useful in this regard and are ideally suited to applications like the international video project which is currently nearing completion. The recent proposal from AECL to host a workshop on this topic is most timely and it is hoped that this initiative will result in a long overdue boost for this area of work.

Acknowledgements

Many Swiss colleagues contribute to the diverse Nagra programmes for R&D and for repository implementation. For the overview presented here valuable contributions to the text were provided by Ian McKinley, Piet Zuidema and Jörg Hadermann.

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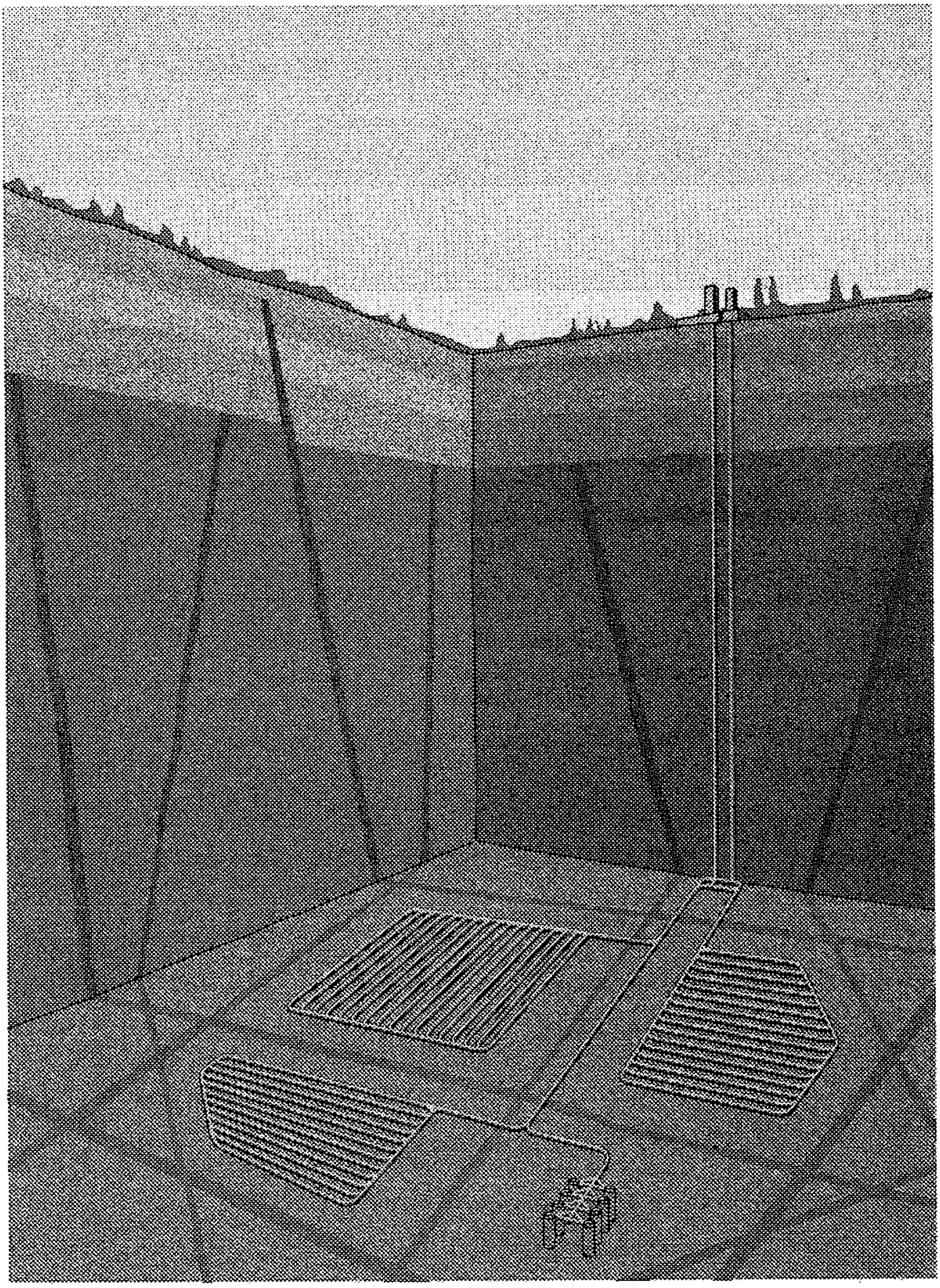
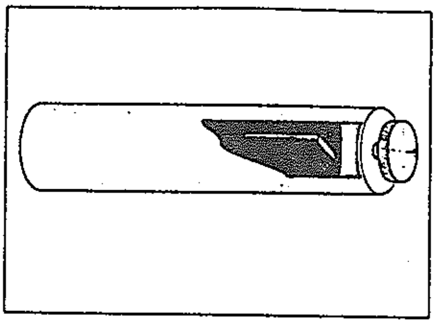


Fig. 1 Schematic illustration of a possible deep repository layout in a crystalline host rock. The layout of the individual disposal panels avoids disturbed zones with potenially enhanced water flow.

Safety barrier system for high-level waste

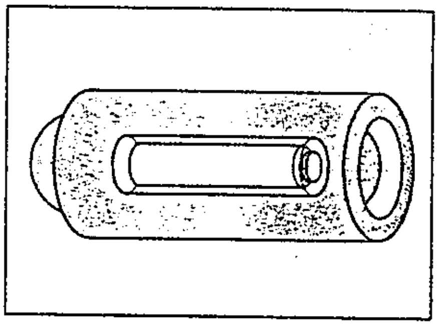
Glass matrix (in steel mould)

- Low corrosion rate of glass
- High resistance to radiation damage
- Homogeneous radionuclide distribution



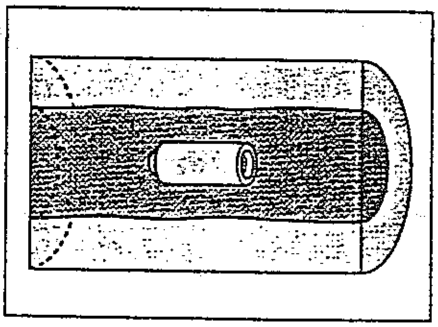
Steel container

- Completely isolates waste for > 1000 years
- Corrosion products act as a chemical buffer
- Corrosion products take up radionuclides



Bentonite backfill

- Long resaturation time
- Low solute transfer rates (Diffusion)
- Retardation of radionuclide transport (Sorption)
- Chemical buffer
- Low radionuclide solubility in leachate
- Colloid filter
- Plasticity (self-healing following physical disturbance)



Geological barriers

Repository zone:
- Low water flux
- Favourable hydrochemistry
- Mechanical stability

Geosphere:
- Retardation of radionuclides (sorption, matrix diffusion)
- Reduction of radionuclide concentration (dilution, radioactive decay)
- Physical protection of the engineered barriers (e.g. from glacial erosion)

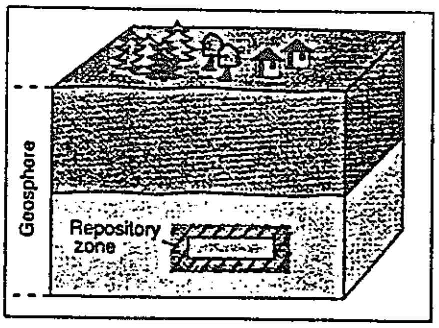


Fig. 2 The safety barrier system for disposal of high-level waste.

UPDATE ON CANADA'S FUEL WASTE MANAGEMENT PROGRAM:

PREPARING FOR THE ENVIRONMENTAL REVIEW OF THE CONCEPT

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SUMMARY

The Canadian Nuclear Fuel Waste Management Program (CNFWMP) was established in 1978 as a joint initiative by the governments of Canada and Ontario. Under the program, AECL is responsible for developing and assessing a concept to dispose of nuclear fuel wastes in plutonic rock of the Canadian Shield. Ontario Hydro has advanced the technologies for interim storage and transportation of used fuel.

The aim of the concept is to isolate the used fuel waste from the biosphere by a series of engineered and natural barriers. During the past fourteen years, AECL has carried out detailed studies on each component of this barrier system. A robust concept has been developed, with options for the choice of materials and designs for the different components.

The disposal concept is being reviewed under the Environmental Assessment and Review Process (EARP). AECL is the "Proponent" for this review, and will submit an Environmental Impact Statement (EIS) describing the disposal concept. The EIS has been written to respond to guidelines issued by the Environmental Assessment Panel responsible for carrying out the review. The future direction of the CNFWMP will depend on the recommendations of the Panel and on the resulting governmental decisions on the appropriate next steps.

If the concept review is completed by 1996, as currently expected, and the concept is approved, the many steps that would be involved with siting and construction of a disposal facility, mean that disposal would not begin before about 2025.

1 Introduction

Responsible industrial societies have managed nuclear fuel waste with a degree of care and consideration for protection of human health and the environment not generally applied to other wastes. From the very beginning, the nuclear industry has recognized the hazardous nature of its waste and the need to manage the attendant risk.

The volume of nuclear fuel waste in Canada is relatively small. The used fuel is being safely stored, and many years of experience have been accumulated with pool storage and dry storage systems. Supporting R&D indicates that these practices can be safely continued for many decades to come[1,2].

Nonetheless, recognizing the need to provide for long-term safe management, the governments of Canada and Ontario, in 1978, decided to establish the Canadian Nuclear Fuel Waste Management Program (CNFWMP). The objective of the program is to investigate the safety and acceptability of a concept for the disposal of nuclear fuel waste in an underground vault constructed in intrusive igneous (plutonic) rock[3]. Disposal is defined as a permanent method of waste management in which there is no intention of retrieving or handling the waste in the future. The incentive for disposal is based on the ethical principle that we, as the principal beneficiaries of the energy generated by nuclear power plants, should assume, to the extent possible, the burden for managing the waste produced in generating that energy. This implies:

- providing the financial resources for managing the waste. Canada's nuclear utilities have already incorporated charges into their rate base to accrue funds for nuclear fuel waste disposal; and

- developing a waste management strategy that ensures the long-term protection of human health and the environment, and that, to the extent possible, does not rely on long-term institutional controls for safety.

In 1981, the two governments reaffirmed their commitment to the program, but announced that a disposal site selection process would not be started until the concept had been reviewed and accepted[4]. Thus, the R&D program and concept development have been carried out on a generic basis rather than a specific project basis.

Participants in the program have included AECL, the lead agency for research on disposal; Ontario Hydro, which has advanced the technologies for storage and transportation as well as contributing to the R&D on disposal; Energy, Mines and Resources (EMR) Canada; Environment Canada; scientists at Canadian universities; and consultants in the private sector. AECL's activities are currently cofunded by AECL and Ontario Hydro through the CANDU Owners' Group (COG).

2 The Disposal Concept

The concept is based on disposal in plutonic rock of the Canadian Shield, which extends over a large part of Canada from Labrador to Alberta. A series of engineered and natural barriers will isolate the nuclear fuel waste from the biosphere. The main elements of the concept include: enclosing the nuclear fuel waste in corrosion-resistant containers designed to have a minimum life-time of 500 years; emplacing these containers in a vault excavated (nominally) 500-1000 m deep in plutonic rock of the Canadian Shield; using buffer materials around the containers to retard the flow of water and radioactive materials, and using seals and buffer material to backfill the vault, access shafts and tunnels[5]. During the past fourteen years, AECL has carried out detailed studies on this multiple-barrier system. The objective has been to develop a concept with flexibility in the choice of methods, materials, and designs for the components of the disposal system. The approach has focused on ensuring that the system as a whole meets safety standards by a large margin.

The choice of methods, materials, and designs for an actual disposal system will ultimately be made on the basis of performance taking into account the characteristics of the specific site on which the facility is to be developed, availability, cost, and practicality. They could include, for example,

- the form of the waste - used fuel bundles or glass;
- the disposal container material - titanium alloy, copper, or other durable material;
- the container design;
- the composition of materials used for the buffer, backfill, and seals;
- the excavation method - blasting or boring;
- the depth, geometry, and the number of levels of the vault;
- the size and shape of the excavated openings; and
- the location of the waste containers - within disposal rooms or in boreholes in the floor of the rooms.

These choices will not be made until a site for a vault has been selected.

3 <u>Environmental Review</u>

The initiating department, Energy, Mines and Resources (EMR)(now Natural Resources) referred the concept for review under the Environmental Assessment and Review Process (EARP) in 1988. As the "Proponent" for this review, AECL will submit an Environmental Impact Statement (EIS) describing the concept. The Environmental Assessment Panel responsible for carrying out the review is chaired by Mr. Blair Seaborn. The Panel has appointed a Scientific Review Group (SRG), chaired by Professor Raymond Price and composed of eminent scientists from a variety of relevant disciplines, to assist it in judging the technical validity and acceptability of the disposal concept. The Federal Environmental Assessment Review Office (FEARO) provides administrative support.

The Panel will review AECL's concept, along with a broad range of nuclear fuel waste management issues. These include the criteria for determining safety and acceptability; the approaches used in handling nuclear fuel waste both in Canada and other countries; the potential social, economic, and environmental effects of waste disposal; and the potential impact of recycling and other processes on waste volume. A general review of other aspects of the nuclear industry, such as energy policy and reactor operation and safety, is specifically excluded from the Panel's review.

All federal departments with a relevant interest in the concept are expected to participate in the review process. These include the Atomic Energy Control Board (AECB), EMR, Environment, Health and Welfare, and Transport Canada. EMR has assembled a team to review the results of AECL's R&D program and

Environment Canada has assembled two teams of experts to review in detail how well the concept protects the environment.

When the EARP review is concluded, the Panel will make recommendations as to the acceptability of the concept and the course of future action regarding nuclear fuel waste disposal. Government decisions will then follow.

FEARO organized a series of "Open Houses" in the spring of 1990 to inform interested parties, not directly connected with the nuclear industry or with the scientific review process, about how they could take part in the review. "Scoping Hearings" took place in the autumn of 1990 to identify issues of concern, and to assist the Panel in setting guidelines for the EIS. One hundred and thirty participants made presentations, including government departments, scientific and business organizations, special interest groups, and private individuals. Among the major issues raised were arguments for and against storage as compared with disposal, the adequacy of the regulatory criteria, and monitoring the performance of the disposal vault. Aboriginal land claims affect much of the land where a disposal vault could be sited. In view of this, an aboriginal representative was added to the Panel.

In June 1991 the Panel issued draft EIS guidelines for comment. Over thirty different groups and individuals submitted comment. The final guidelines were issued in March of 1992[6]. Since then a major effort has been under way within the program to prepare the EIS and supporting documentation describing the program. The EIS and nine Primary References are being written to provide a complete description of the concept and the technology that has been developed over the past 15 years. The EIS also provides additional information specifically requested by the Panel. When complete, the EIS and the nine Primary References will comprise some 6000 pages.

Several aspects of the Panel's review are unique or unusual[7]:

- a concept for disposal rather than a site- and design-specific project is being submitted for review;

- choices are being called for on matters important primarily to future generations;

- the primary purpose of disposal is to protect human health and the environment should societal controls cease to be effective; and

- flexibility in siting and design criteria must be maintained to allow freedom for informed collective decision-making by the public in a future implementation of the disposal concept.

Although this is a review of a concept as opposed to a site-specific assessment, the Panel guidelines require that the EIS discuss issues that may be important to a future site-specific assessment of an implementation of the concept. AECL will assist the Panel in identifying potential issues for such a future assessment by using hypothetical site and design descriptions to indicate that the concept could be implemented with present technology, that an actual disposal system can be assessed, and that a suitable site can likely be found in Canada.

AECL views the current review as the beginning of a continuing process. As the technology for managing the disposal of nuclear fuel waste is developed and applied to specific sites, further reviews and public consultation and involvement will be needed in the future. Any facility will be subject to rigorous regulatory criteria, and it is anticipated that society will demand that a step-by-step process be followed. Thus, a decision to proceed on the basis of the current review would not commit society irrevocably. A judgement now that the concept is safe and acceptable would represent only the first of a series of decisions between distinct phases of the process.

Each phase should lead to increased confidence in the overall system, thus facilitating decision-making about how and whether to proceed to the next phase. We are currently nearing the end of the first phase - concept development and assessment. If the Panel shares our view that we have adequately developed the concept, and there is a governmental decision to proceed, the next appropriate step would be the start of site-specific activities, beginning with site screening. The sequence of events would be as follows:

- site screening would lead to the selection of one or more sites for detailed characterization based on surface techniques;

- such site characterization studies would lead to a selection of one or more sites for exploratory excavation and more extensive in-ground characterization;

- in-ground characterization could lead to a decision to initiate construction and operation of a disposal vault, possibly beginning with a demonstration phase;

- design, construction and operation of a facility would involve ongoing review, reassessment and recommitment, leading to continued operation and then eventually to a decision to cease operations and decommission;

- decommissioning and post-operational monitoring would ultimately lead to a decision to close and seal the vault.

The process of site screening and of evaluating several sites will likely involve a further ten to fifteen years of work before a commitment would be made to initiate an underground excavation, followed by a further ten years or so of site exploration and characterization before construction could begin. Thus, waste would not be emplaced in a vault before about 2025. By then we would have accumulated many years of site-specific data and a series of increasingly refined evaluations on which to base a decision to begin to emplace waste.

The decision to close and seal the vault would be made on the basis of the accumulated evidence and experience gained throughout the siting, characterization and operational phases, a process extending over close to a century. Only with that decision will disposal based on the concept have definitively been judged as safe and acceptable.

Thus, at the current concept assessment phase of the process, "concept approval" does not mean that definitive responses are available for all technical and social issues, because all such issues will not have been resolved. The concept has been developed specifically to be able to accommodate the different conditions and demands that will be specific to particular sites. Rather, we believe that concept approval represents a judgment that:

- sufficient understanding has been developed to continue with the process, with an expectation that we will eventually reach the end point of sealing a vault; and that

- at the appropriate time we should proceed to the next phase of the program, the beginning of site-specific activities to resolve outstanding issues that can only be resolved on a site-specific basis.

4 Establishing Concept Acceptability

The AECB has set out objectives and criteria for the disposal of radioactive wastes at a specific site in a series of regulatory policy statements[8-10]. The development program we have carried out over the past fifteen years has enabled us to develop the tools and expertise needed to obtain site-specific data and to incorporate this information into a disposal vault design that will meet these regulatory criteria.

To demonstrate this, we intend to establish during the environmental review that

- technology exists to site, design, construct, operate, decommission and close a disposal facility that meets the regulatory requirements for the protection of human health and the environment;

- a methodology is available to evaluate the performance of a disposal system in plutonic rock in terms of regulatory requirements for the protection of human health and the environment; and

- it is likely that a suitable site can be found in Canada.

Because of the governments' requirement that no site be selected prior to review and acceptance of the concept, AECL has not assessed a specific site with its particular body of rock and its particular surface environment. Rather, three related case studies have been performed, each with its own objectives relative to the assessment:

- We have developed a conceptual design of a hypothetical disposal facility. This design has been used to assess engineering feasibility and costs, and to provide information relevant to assessing the potential impacts of disposal[11].

- Ontario Hydro has assessed the short-term preclosure impacts of implementing the conceptual design at a hypothetical site[12]. The objectives were to demonstrate the assessment methods, to determine how sensitive the estimated impacts are to changes in the factors considered, and to indicate the type and magnitude of impacts that could occur.

- We have assessed the long-term postclosure impacts of a hypothetical disposal facility at a hypothetical site having subsurface characteristics derived from information obtained from a field research area[13]. The objectives were to demonstrate the assessment methods we have developed, demonstrate how the assessment methods are used as a design tool to determine design constraints; to establish the relative importance of various design parameters; to determine how sensitive the estimated impacts are to changes in the factors considered; and to show that a disposal system, under hypothetical but realistic conditions, could meet the safety criteria.

Thus, in our development program we have demonstrated our ability to investigate the surface and subsurface characteristics of potential host rock formations, we have demonstrated specific aspects important to the engineering of a disposal system, and we have developed a conceptual design of a hypothetical disposal facility.

Although it is not possible to provide complete full-scale demonstrations of all aspects of a disposal facility without actually building one, our case studies are based on realistic facility and site characteristics, albeit hypothetical, using information obtained from extensive laboratory and field research. The hypothetical disposal facility is technically feasible with available technology or with reasonably achievable developments, as required by the AECB[8], and the characteristics specified for the hypothetical site are, in our opinion, not exceptional.

While many detailed investigations would have to be done at an actual candidate site to establish its suitability for a waste disposal facility, we argue that the requirements of a technically suitable site are likely to exist on the Canadian Shield.

5 Future Direction - The Importance of Proceeding Towards Implementation of The Concept

The future direction of the CNFWMP will depend on the recommendations of the Panel and the resulting governmental decisions as to the appropriate next steps. Ontario Hydro has published a Corporate Reference Plan for Used Fuel[14], and AECL, EMR, and the utilities have initiated discussions to be in a position to proceed with implementation if the concept is approved. Siting activities are clearly important initial activities. These include site screening, characterization of one or more potential sites using surface techniques, and excavation and in-ground characterization of at least one site. Implementation will require design optimization taking into account site specific information, and continuing development of key technologies to support this optimization.

If the Panel recommends acceptance, we believe that it is important that Canada proceed, without delay, to the next step leading towards implementation, for the reasons discussed below.

Environmental Leadership and Reducing the Burden on Future Generations

The incentive for selecting a permanent disposal concept for managing long-lived nuclear fuel waste derives from two fundamental ethical principles:

- the wastes must be managed in such a way that human health and the environment are protected in the short and long-term, and

- as the principal beneficiaries of the energy which gives rise to the waste, our generation should assume to the extent possible, the burden of managing the waste.

These principles underly the objectives, criteria and guidelines that the Atomic Energy Control Board has set for judging the safety and acceptability of radioactive waste disposal[10].

From the outset, the Canadian Nuclear Fuel Waste Management Program was founded on the principle that we have an obligation to protect and avoid burdening future generations. This belief was supported by the conclusions of an ethics workshop conducted in March of 1991 with eight ethicists, social scientists and theologians, including an Aboriginal leader. A report summarizing the proceedings of the workshop has been produced[15]. The recommendations from the workshop, broadly stated, were that:

- the generation that benefits from nuclear power must take responsibility for disposing of the resulting waste;

- no burden should be placed on future generations, but they should have the options of retrieving the waste and of taking remedial action if necessary; and

- decisions ought to involve informed consent from the affected public.

Each of these recommendations is reflected in AECL's concept in some form. Minimizing the burden on future generations means more than simply making financial provisions. It means, to the extent possible, providing the technology to implement disposal and providing it in such a way that future generations retain flexibility in their decision-making.

Since the inception of the nuclear power industry, the industry has shown environmental leadership in managing its radioactive wastes, operating on an essentially closed fuel cycle. The nuclear power industry is the first industry in Canada to have managed and accounted for its full life-cycle costs, including environmental costs. By proceeding with the development of disposal technology the industry will continue to meet its ethical responsibilities and will demonstrate that nuclear power truly is a sustainable source of energy.

Fostering Public Confidence in Nuclear Energy

A second reason for proceeding along the path toward disposal following concept acceptance is to respond to public concerns associated with the use of nuclear energy. Public confidence in the capability of the industry to dispose of nuclear fuel wastes safely is important to maintaining public confidence in nuclear power as an energy source. AECL public opinion research shows that two-thirds of the Canadian public say that nuclear power would be more acceptable if a permanent solution could be found for the disposal of nuclear fuel waste. Therefore, progress needs to continue to be made towards addressing the long-standing public concern about the final disposition of fuel waste.

The need to respond to public concerns has been repeatedly cited in reviews of energy supply and the nuclear industry. In 1980, the Porter Commission established by the government of Ontario to examine electric power planning, concluded that:

> "If progress in high-level nuclear waste disposal R&D, in both the technical sense and the social sense, is not satisfactory by at least 1990... a moratorium should be declared on additional nuclear power stations[16]."

(Professor Porter has since said that he believes this condition has been met.)

In 1988, the parliamentary Standing Committee on Environment and Forestry published a report calling for a moratorium on further construction of nuclear power plants in Canada until a permanent disposal method for used fuel was demonstrated[17]. Whilst this recommendation was rejected by the government, it reflected public concern consistent with the results of public opinion polls.

Later in 1988 the Standing Committee on Energy, Mines and Resources issued their tenth report[18]. The report was favourable towards the nuclear option, but recommended acceleration of the Concept assessment process, specifically "to strengthen public confidence that the longer-term issue of disposal is being satisfactorily resolved."

At the World Energy Conference in Madrid in 1992 September, it was evident that while nuclear energy is well-positioned to play an important role in meeting the world's growing demand for energy in an environmentally-sustainable manner, public confidence is a key issue. Waste management is an important part of this issue.

To quote from one paper, "Electricity, the Environment, and Sustainable World Development," submitted at one of the plenary sessions:

> "Though nuclear energy has the advantage that it emits none of the atmospheric pollutants of concern with fossile fuel technologies, the fission reaction does generate long-lived radioactive wastes, ultimate disposal of which is extremely controversial[19]."

We believe that the public process that has been followed in developing the concept, the scientific scrutiny to which the concept will have been subjected, governmental acceptance based on the recommendation of the review Panel, followed by continued progress leading towards implementation, would increase public confidence in the nuclear industry and assist in resolving concerns and controversy regarding nuclear energy.

Forestalling Inaction by Default

The third reason is that, unless there is a clear intention to implement the concept if it is accepted as a result of the Panel's review, there is a great potential for the review process to fail to lead to a clear commitment to action. Throughout the Panel's review, AECL, as the proponent, needs to continue work to address outstanding issues and continue long-term experiments to demonstrate the technology, and the industry needs to demonstrate a firm intention to proceed with disposing of its wastes in an environmentally sound manner if the concept is accepted as a result of the Panel's review. The review process, as is evident from the reviews of other projects - the Pearson airport, and uranium mining projects, for instance - may be long and drawn out. We need to provide the background for the Panel and the Government to have confidence in a feasible and logical next step. Without such an impetus towards implementing the disposal concept, Canada could fail to take any action by default.

Preserving the Knowledge Base

This risk of inaction by default leads to the last reason for proceeding: to meet our ethical responsibilities to future generations it is important that we preserve the knowledge base that has been generated from the investment that has been made to date. A great deal of technical knowledge has been developed in the course of this program. Tapping this expertise will be essential to successful implementation. While much of the information has been documented, of even greater importance is the ability to interpret the results of tests, measurements and other observations. To give just one example, Canada now has a unique capability to develop a conceptual model of groundwater movement deep in plutonic rock that stands up to comparison with field observations. This capability must be used if it is to continue to be available.

If the review is completed by 1996, as currently expected, and if the concept is approved, disposal could not begin before about 2025. Although this date appears to be far in the future, the many activities associated with siting a facility, optimizing the details of the concept, and public consultation and review processes will require that amount of time. By 2025, Canada's current generating stations will have produced some four million bundles or so of used fuel, all of which will be stored. More importantly most of the current plants (Pickering, Bruce, Pt. Lepreau, Gentilly) will have reached or will be approaching the end of their design lifetimes. It is therefore important that we proceed with the next steps leading towards implementation without delay, once the concept has been accepted.

6 Conclusion

AECL believes that it has developed a robust and flexible concept for disposal of nuclear fuel waste that will meet the regulatory requirements of Canada. We are continuing research and development work to ensure the public and the industry has as much confidence as possible in the safety of the concept and in the feasibility of implementing it. The process for a federal environmental review of the concept is well under way.

The review of a concept as opposed to a site- and design-specific project requires focusing on whether it is appropriate to proceed with the first phase of implementation. We believe that we have reached the stage in the CNFWMP where the greatest benefit will result if activities proceed on a site-specific basis.

We are entering a very public process. Our experience has shown us that such processes are not easy for the nuclear industry. We are confident of our ability to provide a thorough and convincing EIS, which should lead the Panel to recommend that we proceed to the next phase of the process leading toward disposal. Our confidence is founded on the strength and depth of our technical program and a well-founded public consultation program. Building on this foundation, we can begin to address the many demands that this review will place on us.

Given a positive outcome of the review, it is important that the industry take the next steps towards implementing the concept to meet our ethical responsibility to future generations, to secure public acceptance of nuclear energy as a sustainable energy option, to forestall inaction by default, and to maintain the capability to implement the concept.

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Current Status and Future Plan of Research and Development program related to geological disposal at CEN•SCK

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General

The studies at CEN•SCK related to the R&D on geological disposal are performed within the research unit "waste and disposal" which also deals with the characterization of waste packages and their compatibility with the host clay.

Enhancing the scientific output remains one of the main objectives of our programmes; in this respect it is our purpose to still broaden our research activities from the domestic to the more international scientific and technical markets.

An overall global view of the in situ test programme developed from the underground research facility (URF) is given by figure 1. A proposal for extension is represented in dotted lines (see "Praclay" demonstration test in point 2.2).

Recent developments on CEN•SCK efforts in the programme of ONDRAF/NIRAS, the Belgian Waste Management Authority

Actions launched before 1990

The following tests are being continued:
- the in situ corrosion and leaching test set-ups,
- the combined radiation/heating test CERBERUS,
- the in situ migration experiments,
- the geotechnical survey on the gallery structures and their surrounding clay.

The main achievements until mid 93 can be summarized as follows:

Corrosion

The CEN•SCK has developed in the early 80's a screening programme on the corrosion behaviour of candidate container materials and conditioned waste forms. The specimens were selected out of the various candidate materials studied within the European Community and/or for specific Belgian interests and emplace in the URF in view of getting conditions close to those prevailing in a final repository.

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The interactions between the overpack materials and the host formation were lasting for periods up to 7 years. Such tests help in establishing a first estimate of the corrosion resistance required to ensure confinement by the overpacks for periods of at least 500 to 1000 years.

Many samples retrieved from these in situ test loops have been analyzed.

The following average corrosion rates for C-steel were determined (in $\mu m.year^{-1}$): 1.81 at 16°C, 7.68 at 90°C, and 8.59 at 170°C.

Recent analyses confirm the expected presence of thin passive layers on the corrosion resistant materials.

From the mass loss measurement of glass specimens, it was found that at 16°C the average corrosion rate is below 0.25 $\mu m.year^{-1}$, and at 90°C between 6 to 100 $\mu m.year^{-1}$, depending on the glass composition.

New test loops are planned to be installed still this year.

Near field

The CERBERUS-test, aiming at simulating the near-field effects of a HLW canister in an argillaceous environment (using ^{60}Co sources), is continued and the first medium term effects could be identified. The extensive instrumentation program of the clay includes the monitoring of temperature, doses, interstitial and total pressure as well as pH/Eh.

The set-up of the Cerberus mock-up also gives the opportunity to test the behaviour of backfill material, waste matrix, overpack and canister material under near-field conditions, by sampling and analyzing the clay and the components at the end of the test.

Several tests were performed to investigate the permeability of the enveloping clay and its evolution in function of time after the steady state of temperature and pore water pressure was reached. A set-up for the detection of radiolysis gas was conceived, optimized (detection limit) and applied. During a first measuring campaign, a concentration of 0.52 $\mu g\ H_2/Kg\ H_2O$ was determined.

Migration (radionuclides)

Migration tests are carried out to study the migration of actinides, fission products, non-sorbed species and recently also the mobility of the dissolved organic material. Large scale 3-D experiments with tritiated water and iodide are designed and installed from the underground research facility to confirm the anisotropy of hydraulic parameters and to validate the migration model. The concentration of the migrating radionuclides measured in the interstitial water agree very well with the predictions calculated using the MICOF-computer code (INTRAVAL exercise).

A second large scale in situ experiment in a vertical and a horizontal piezonest was started in June 1992, using $^{125}I^-$ as tracer. Again, the concentration values measured up to now agree quite well with the predictions.

An array of three other piezonests for a third large scale in situ 3D injection experiment with tritiated water is successfully installed from the underground facility and will be operational next year.

Two Am/Tc experiments were installed. The first experiment just started is located near by the Cerberus experiment. The other one, to be used as a reference, is located in the same horizon but at a place where it can not be affected by any radiation or temperature effect.

Geomechanics

Various civil engineering works and underground experiments have been carried out to assess the technical feasibility to build a high-level waste (HLW) repository in a plastic clay formation. Most of these studies (long term behaviour, damaged zone, geotechnical campaign,...) have been performed from 1980 to 1989 in connection with the different stages of construction on site. The mining capabilities in clay were demonstrated, using conventional techniques at full scale.

The last two years many studies have also been launched, allowing to develop analytical tools and/or benchmark exercises based on field measurements.

No recent geomechanical in situ experiment are to be mentioned. Only long term survey is running. The good mechanical behaviour of the shotcreted front end of the test drift gallery has to be noticed.

Recently started actions

New actions have been started or re-launched again since 1991 related to:
- the hydrogeological modelling,
- the migration of gas in clay,
- the "PRACLAY" demonstration test.

Hydrogeology

Besides the periodic water level measurements in the regional hydrological observation well network around the site, the main efforts were devoted the last 2 years to a critical reviewing of the previous regional hydrological modelling and to a detailed overview of all available in situ data.

A critical discussion has been conducted by commissioning several evaluations and leads to an improvement to the model.

In a first step, the improved version of the former Newsam model will be used to run new simulations.

Migration (gas)

The migration studies have been extended to the permeation of hydrogen gas. The MEGAS project (Modelling and experiments for gas migration), part of the CEC umbrella project "PEGASUS", started in 1991.

A methodology similar to the one of the radionuclide's migration programme was followed. A model is developed to design, simulate and interpret support laboratory tests.

The reaction capacity appears to be negligible for undisturbed Boom clay and rather limited for oxidized clay. Other laboratory experiments (diffusion, gas breakthrough) indicate that the flow of gas displaces only a very small fraction of the pore water. Two gas flow mechanisms seem to occur; from a certain pressure level, a flow through preferential pathways becomes predominant with regard to the expected two phase flow observed only at low pressure levels.

Four piezometers in a 3D configuration have been installed from the underground laboratory for gas injection experiments. The experiment serves as validation test for the TOPAZ code developed by INTERA.

PRACLAY

First preliminary actions were launched the last two years for the demonstration test "PRACLAY", intended to simulate and investigate the thermal conditions in and around a HLW-disposal gallery (30 m long, 2m in diameter). The reference concept for this test is the centrally installed heat producing waste, inside a steel support tube, surrounded by a bentonite-based backfill material.

This project, managed by ONDRAF/NIRAS, is part of CEC's R&D programme on radioactive waste management and disposal (Part B). Construction and emplacement techniques on semi-industrial scale (tunnel, lining, shroud, backfill) will be demonstrated. The installation and the operation of this experiment is scheduled to last until about 2000.

The heat dissipation of the radioactive waste will be simulated by heating cables (linear power of 450 W/m) placed inside the central tube. A concept that allows the replacement of the heating cables, in case of failure, was developed and is now tested.

The main efforts of CEN•SCK are concentrated on the instrumentation issues of the clay environment (calibration of devices, procedure of installation) and of the concrete lining of the PRACLAY tunnel. An extensive literature review evaluating similar set-up's and experiences was first carried out.

Recent developments in CEN•SCK's efforts in the international context

The collaboration with ANDRA, started in 1984 with geotechnical testing and extended from 1987 to full scale tests (gallery lined with sliding rib systems, CACTUS tests), is pursued in the 90's.

In the framework of the actual CEC programme on demonstration and pilot facilities for waste disposal, two new projects, managed by ANDRA but involving important in situ developments from the HADES URF at Mol, were launched during the second semester of 1992: the ARCHIMEDES-argile project and the PHEBUS test, both dealing with site characterization.

Furthermore, the study on clay-based backfill materials was resumed with European partners to develop a model for the simulation of the behaviour of such material in unsaturated conditions. An in-situ test was designed together with the modellers to validate this model (BACCHUS-2) and installed from the URF.

A new thermo-mechanical test (ATLAS), similar to the BACCHUS and CACTUS experiments but designed according to the "in-gallery" or "axial" concept, was installed in 1992. It is aimed at validating model developments in the framework of the INTERCLAY II benchmark exeroise.

These four main experiments are hereafter shortly described.

ARCHIMEDES

Besides the hydrogeological characterization of the Boom clay and surrounding geologies, this project is intended to understand the mechanisms governing the acquisition and the regulation of the water chemistry in a clay environment and to validate the geochemical models describing the behaviour of the radioelements in clay on basis of representative field data for the key chemical parameters.

It proceeds along four main activities: fluid and solid sampling at the field, laboratory analyses, microbial studies and fluid-rock interaction modelling.

Up to now, the undisturbed and representative fluid and solid samples needed by the partners involved in the project were taken from our underground facility; the adequacy and performance of sampling techniques are being tested and compared.

Two core drillings (20 and 15 m long) have been achieved successfully under sterile and anaerobic conditions. The holes were equipped with sterile piezometers making 10 filter screens available for sampling pure and biologically not contaminated clay water for the geochemical and the microbial analyses.

Frozen core samples incorporating the interstitial water were taken in order to avoid any risk of exchange of hydrogen and oxygen isotopes with the atmosphere. Other experimental piezometers have been installed to investigate the pore water chemistry (Ph/Eh, cation exchange capacity, dialysis).

PHEBUS

This test is aimed at investigating the water exchanges between a clay formation and a ventilated underground structure. It consists in two main phases: the understanding of the water exchange process using a mock-up where specimens of clay are tested and the modelling of such a process in parallel with a small scale test in the Boom clay, to be installed in 1993 from the Hades URF.

First design studies for the PHEBUS experiment were performed using existing computer codes. An overview was carried out to select the most adequate instrumentation for total and pore-water pressure as well as moisture content, considering experimental conditions and limitations. In particular, the design of a collimated neutron probe for the investigation of a limited volume of clay has to be mentioned.

BACCHUS 2

In cooperation with European partners (SP, UK, F) computer codes for the hydro-mechanical and thermo-mechanical behaviour of unsaturated clay based backfill material have been written and verified. These codes will now be applied to the first results of laboratory experiments measuring the progress of a hydration front, monitored by X-ray tomography, and by measuring the influence of the applied suction on the hydro-mechanical properties of backfill materials based on Boom clay.

An in situ test, called "BACCHUS 2", was designed for the demonstration of the applicability, on an industrial scale, of a backfill material composed of high density Boom clay pellets mixed with Boom clay powder. It was installed in clay in June 1993 after the retrieval of the test cell of the former Bacchus experiment for expertise.

ATLAS

Atlas is an in situ experiment which consists in the simulation, on a reduced scale, of the thermal output of a waste canister emplaced along the axis of a gallery in a clay medium and in the monitoring of the clay behaviour. The field data recorded (temperature and pressure evolution) will allow for the validation, with regard to this specific application, of the models proposed by the different partners involved in the benchmark exercise INTERCLAY II.

The test is already installed and a sufficiently long time period has to be allowed for the restoration of the initial ground conditions. The heating phase started mid 93.

Future perspectives

Beside the further development of above-mentioned experiments, new issues will be addressed in the future, taking into account the increaded need to provide more reliable data, methods and tools in view of getting acceptance about the geological disposal and the need to extent the research in various relevant fields already identified but not yet fully investigated.

The site characterization programme could be e.g extended to the direct impact of shaft sinking and tunnelling in a sedimentary series.

New large scale demonstration tests will be developed to confirm the mechanical, chemical and physical interactions between the various man-made repository components (waste package, backfilling, lining,...) and the natural but altered near-field (argillaceous host rock altered by excavation, oxidation, heat and radiation). The research now running clearly shows the need of complementary multi-approach investigations of the importance of the near-field effects and changes on the source/release term.

The disposal potentials for a wider spectrum of waste types (spent fuel, Mox-fuel, waste from site restoration,...) will also be considered in more detail in the near future, together with potential retrievability of waste.

SCK•CEN

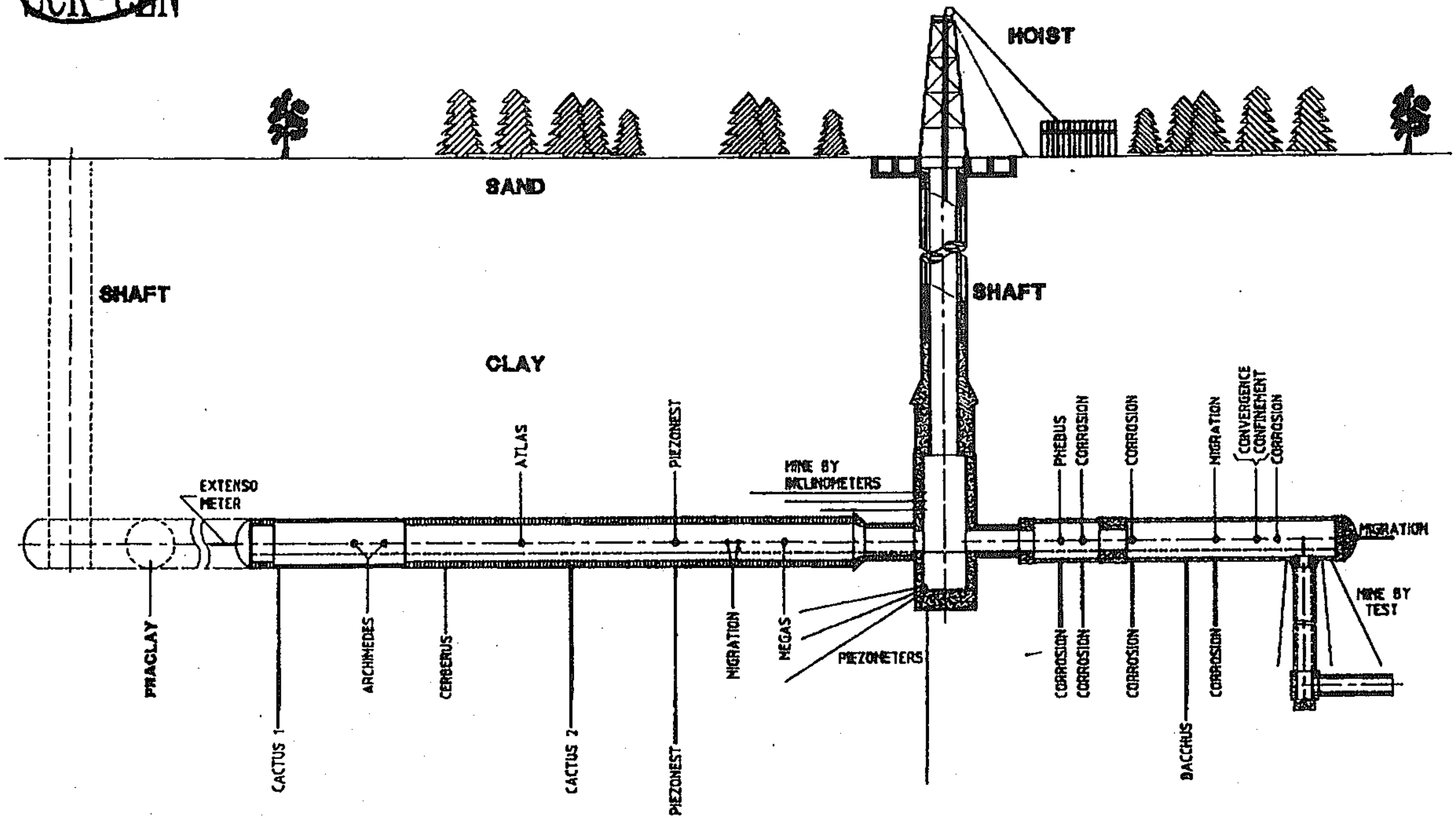

Fig. 1 : In situ test programme HADES

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Current Status and Future Plans of R&D on Geological Disposal of HLW in Japan

Noriaki Sasaki[1)]

1. Background

The Power Reactor and Nuclear Fuel Development Corporation (PNC) has being conducted research and development on the geological disposal of high-level radioactive waste (HLW), as the leading organization, in accordance with the overall HLW management program defined by Atomic Energy Commission (AEC) of Japan(1,2). The responsibility of PNC is to ensure smooth progress of research and development project and to carry out studies of geological environment. The role of the Japanese government is to take overall responsibilities for appropriate and steady implementations of the program, as well as enacting any laws or policies required. On the other hand, electricity supply utilities are responsible to secure necessary funds for disposal, and in accordance with their role as waste producers, they are expected to cooperate even at the stage of research and development.

Fundamental features of research and development of PNC carried out at this stage are as follows;

(1) Site generic research and development,
(2) To establish scientific and technical bases of geological isolation of HLW in Japan,
(3) About 15 years program from 1989 with documentation of progress reports,
(4) Approach from near-field to far-field.

Based on these fundamental features, research and development has been emphasized in the following three subjects;

(1) Geological environment studies,
(2) Disposal technology development,
(3) Performance assessment studies.

PNC summarized the findings obtained by 1991 on these three subjects, and submitted a document (H3 Report) in September 1992 as the first progress report(3). H3 Report is the first and comprehensive technical report on geological disposal of HLW in Japan, and provides information for the public to find out the current status of the research and development.

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2. Overview of H3 Report

(1) Geological environment studies

From the data accumulated in the field of geoscientific research, information which is important from the point of the geological disposal was surveyed and overviewed. Areas of information are the Japanese geology, major rocks, groundwater and alteration products. The present stage of development in survey techniques and equipment for examining the geological environment was also summarized. In the field of major natural phenomena related to the stability of the geological environment, characteristics were examined on the subjects of earthquake activities, faulting, uplift/erosion, volcanic activities, climate change/sea-level change.

It has been concluded in the report that;

- More information of geological environment must be gathered,
- Since major natural phenomena related to the stability of the geological environment are regional, assessment methods and techniques should be developed for more understanding of the regional characteristics.

(2) Disposal technology development

The basic concepts of engineered barriers and repository, candidate materials of engineered barriers, and repository technology (design, construction, operation and sealing) are studied and evaluated. Figures 1 and 2 show examples of engineered barrier design and a full-scale carbon steel overpack made through design and fabrication studies, respectively.

It is concluded in the report that the present technologies are available and applicable to the design, fabrication, construction of engineered barriers and repository. It is recommended, however, to make efforts to develop more reliable technologies.

(3) Performance assessment studies

Concerning to the multibarrier system performance based on groundwater scenario, the followings are mainly considered and analysed;

- Performance assessment methods of multibarrier system,
- Model development of geological environment,
- Evolution of groundwater chemistry,
- Near-field conditions (thermal, hydrological, chemical, mechanical),
- Corrosion of carbon steel overpack,
- Long-term stability of bentonite,
- Nuclide migration in the engineered barriers,
- Nuclide migration in natural barrier.

Figure 3 shows the analytical model chain used in the report to analyse the performance of multibarrier system. The most important conclusions from the analyses are as follows.

- The performance of multibarrier will be retained for long time and therefore geological disposal will be safe, if engineered barriers and repository are designed appropriately in accordance with the geological conditions.
- Near-field is expected to have high nuclide retardation effect.

3. Results of Reviews and Evaluation for H3 Report by AEC of Japan

The AEC of Japan issued the report on the results of reviews and evaluation for H3 in July 1993 (4). In this report, it is concluded that the research and development on geological disposal of HLW has been conducted according to the policies of the Japanese government, and progressed appropriately and steadily. In addition to this evaluation, the followings are recommended as the important items to be studied or considered for the next stage.

(1) Geological environment studies
- Improvement of characterization technology and equipment
- Aquisition of reliable data
- Studies on effects of natural events
- Development of natural event characterization methods of the site

(2) Disposal technology development
- Development of more reliable engineered barriers
- Development of new materials for overpack and buffer
- Comprehensive technology development of design, construction and operation of repository
- Tests under more severe conditions on safety

(3) Performance assessment studies
- Assessment of near-field performance
- Improvement and validation of models
- Expansion of data base
- Natural analogue study

(4) Construction of underground research laboratories

(5) International cooperation

Research and development in PNC is conducted based on these recommendations to achieve the next milestone.

4. Overall Procedures and Schedule for Implementing Geological Disposal

The AEC of Japan issued overall program for high-level radioactive waste disposal in August 1992 (2). According to this program, procedures for implementing geological disposal are as follows;
- Establishing the Steering Committee on High Level Radioactive Waste Project (SHP) on 28th of May, 1993,
- Establishing the implementing organization around the year 2000,
- Site selection with local acceptance, site characterization, demonstration of disposal technologies, design of repository and application for licensing by the implementing organization,
- Safety investigations for granting licenses conducted by the government,

- Start of repository operations some time after the year 2030, but no later than the mid-2040's.

The SHP's activities are assigned as the followings;

- Planning of implementation of disposal,
- Consideration of implementation organization form,
- Consideration of cost collection,
- Public relations,
- Consideration of regislation,
- Others.

5. Future Plans of Research and Development in PNC

The second progress report of PNC is scheduled to be submitted before the year 2000. The main target is the detailed analysis of the near-field(including engineered barriers) performance. This is based on the possible importance of near-field to the total system performance, which is one of key massages of H3 Report ;

The importance of near-field can be recognized as the followings;

- Near-field can be characterized in detail.
- Near-field performance can be assessed with high reliability.
- Near-field has possibly high nuclide retardation performance.

When these are correct, repository will be quite robust.

Figure 4 shows the schedule of research and development in PNC. In Tokai Works, a new research facility "Geological Isolation Research Facility" was constructed. It started the operation in October 1993 to study mainly engineered barrier materials, near-field phenomena and system performance. In Kamaishi mine, second phase experiments started in April 1993. Results from these and other activities will be contributed to the second progress reports.

Independent of the experiments carried out in existing shafts and drifts such as Kamaishi and Tono mines, there is a plan to construct deep underground research facilities. Considering the variety of geological environments in Japan, it is desirable to have more than one such research facility(2). Horonobe in Hokkaido is a candidate site for such a facility in a sedimentary environment.

Major research and development items for the next stage are as follows.

(1) Geological environment studies

- Near-field groundwater flow and chemistry
- Near-field mass transport
- Characterization of excavation damaged zone
- Site evaluation methods and tools on natural event

(2) Disposal technology development

- Overpack material alternatives
- Additives in buffer
- Large scale engineered barrier tests
- Sealing/grouting technology

- Repository design study
- Effects of seismism on engineered barrier system

(3) Performance assessment
- Data base development
- Model development and Validation
- Sensitivity analysis
- Variation case studies of groundwater scenario
- Safety criteria/standards

6. Remarks

The PNC's second progress report is considered to be quite important in overall procedures and schedule of geological disposal in Japan. The Japanese government, utilities and citizens have much attention to the progress of the research and development and the establishment of implementing organization. The Japanese government plans to set up a special committee to review the progress report.

The research and development of geological disposal is extremely interdisciplinary, and is requested the participation of experts from universities, research institutes and others. International cooperation is also important. The research and development will be continued extensively under these considerations to achieve the next milestone in Japan.

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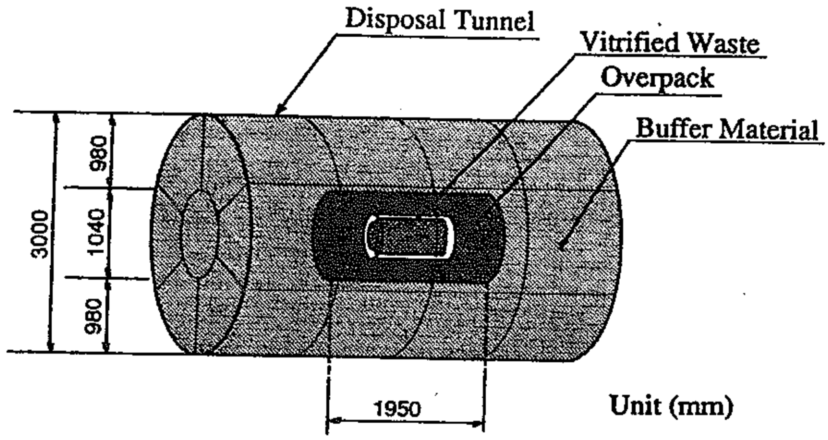
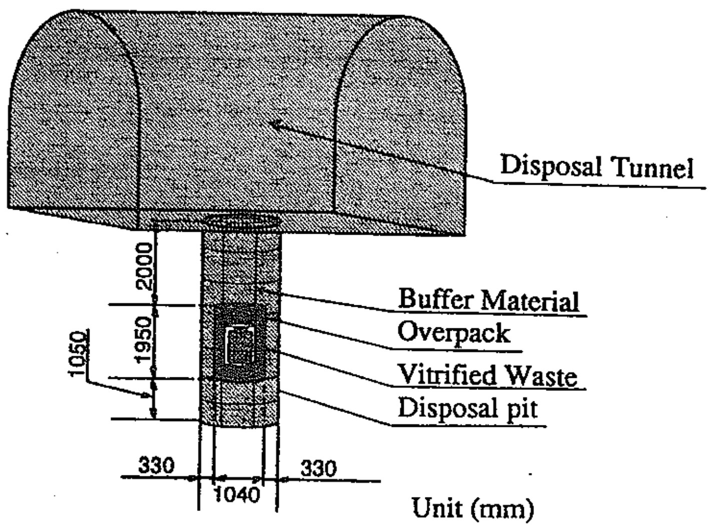


(a) Tunnel Disposal



(b) Pit Disposal

Fig. 1 Examples of Design of the Engineered Barriers

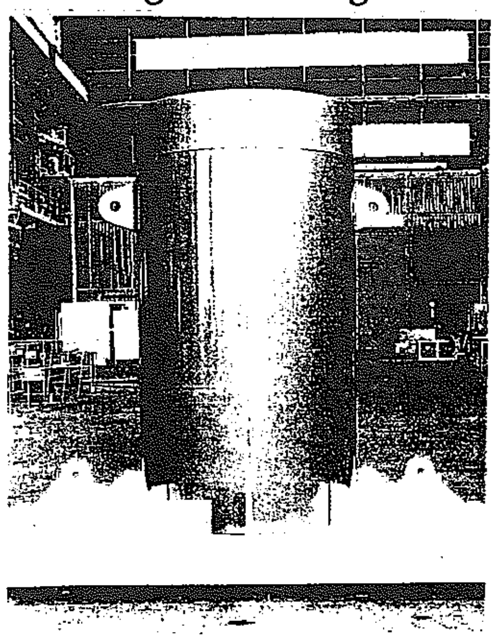


Fig. 2 Carbon Steel Overpack(1.09mϕ, 1.95mH, weight 11.4t)

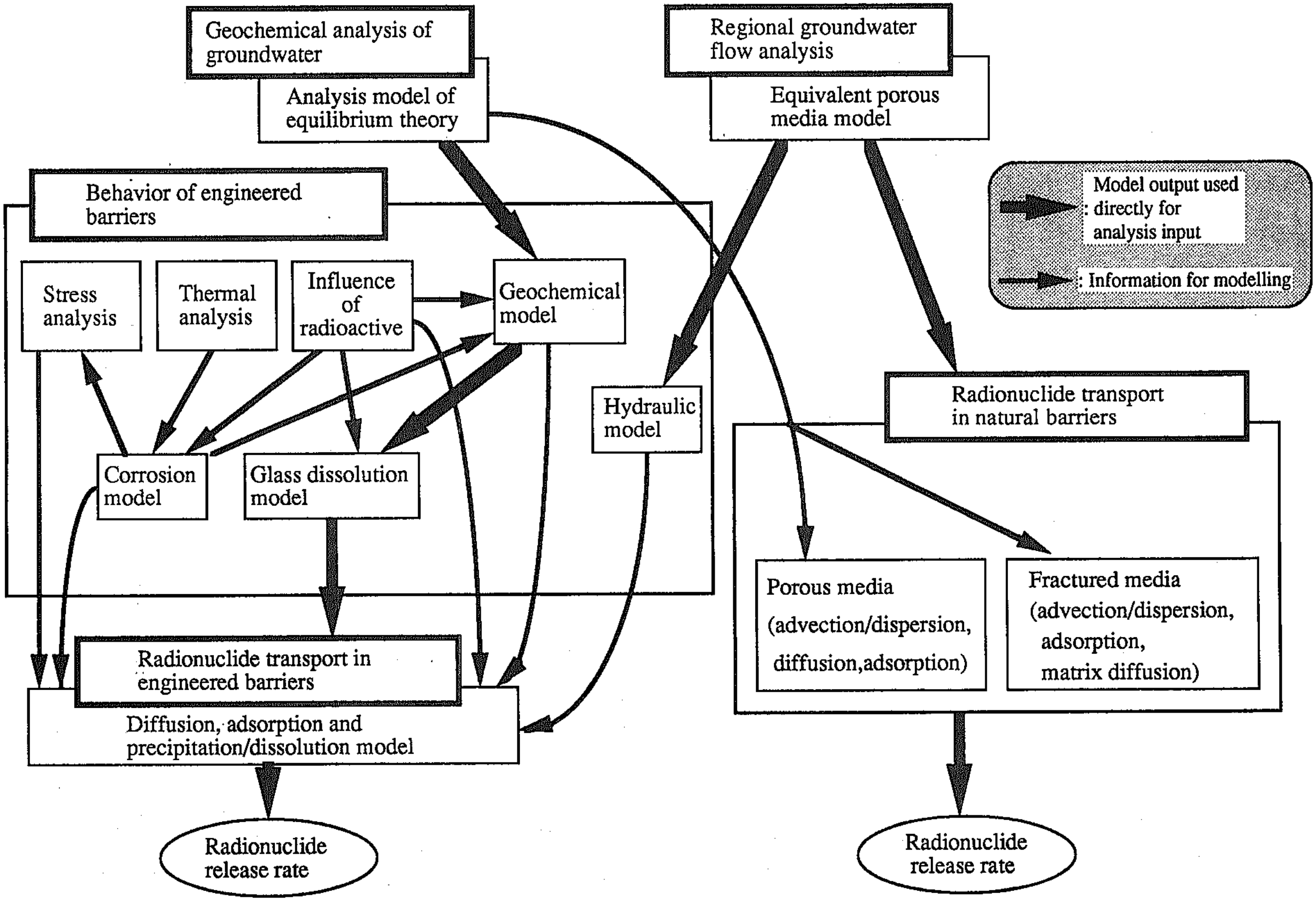


Fig. 3 Model Chain for Analysis of Performance of Multibarrier System(3)

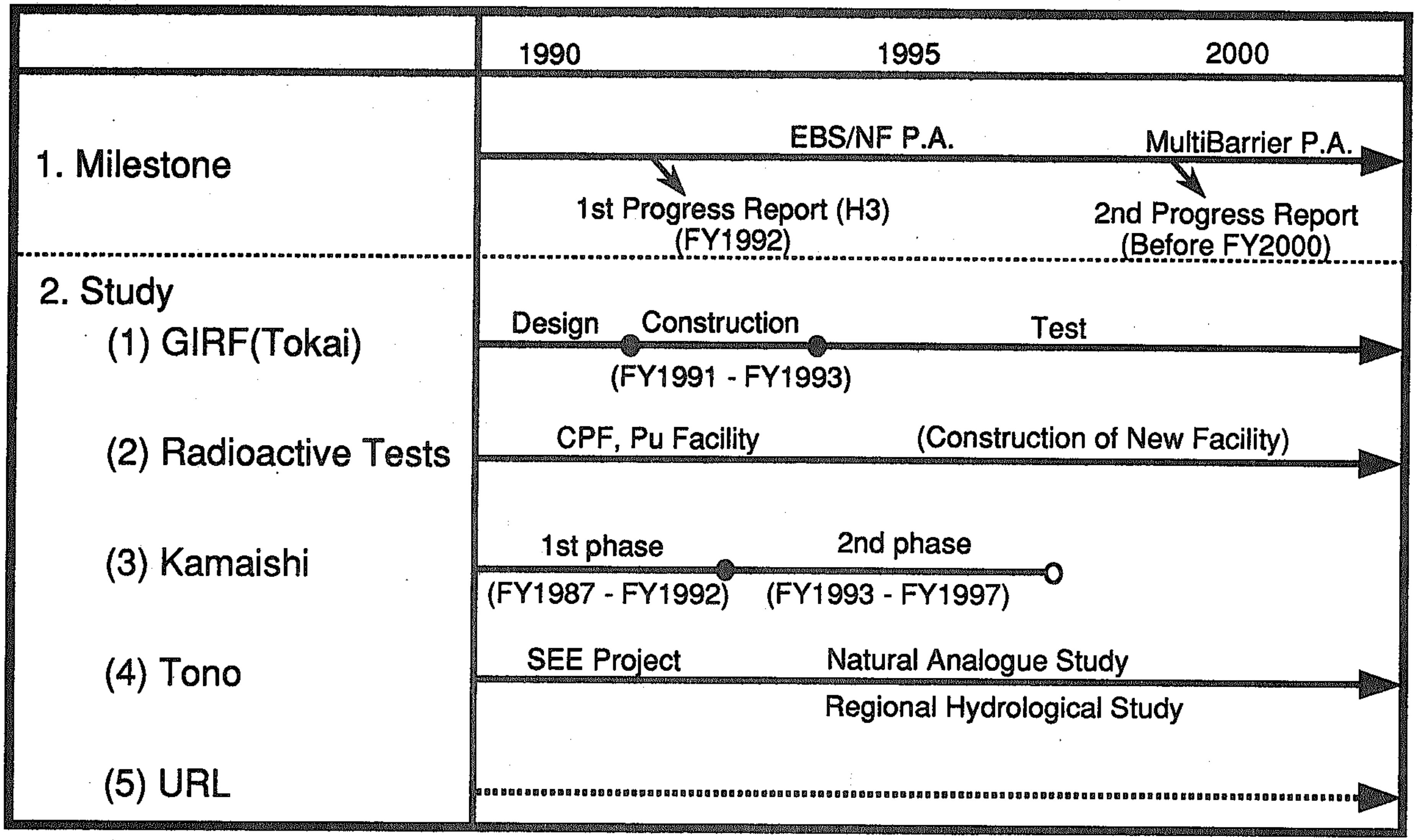


Fig. 4 Schedule of the Research and Development in PNC

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The Origins of Our Mark

The peacock *(kujaku)*, since olden times said to be an eater of a poisonous snake, was deified as a *"Kujaku-Myo-Oh"* (envoy of a god). This *"Kujaku-Myo-Oh"* was worshipped for its ability to get rid of all poison, bring rain and stop the rain.

The pattern on the peacock's feathers has been designed into the symbol of geological disposal. The layered circles of this design represent the system of multiple barriers. This symbolizes the radioactivity of high-level radioactive waste confined by means of these multiple barriers system, so that it will not affect the human environment, even if these barriers come in contact with water.